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RADIOLOGICAL SURVEILLANCE STUDY AT THE HADDAM NECK PWR NUCLEAR POWER STATION

OFFICE OF RADIATION PROGRAMS
U.S. ENVIRONMENTAL PROTECTION AGENCY
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RADIOLOGICAL SURVEILLANCE STUDY AT THE HADDAM NECK PWR NUCLEAR POWER STATION

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FOREWORD

The Office of Radiation Programs of the Environmental Protection Agency carries out a national program designed to evaluate population exposure to ionizing and non-ionizing radiation and to promote development of controls necessary to protect public health and safety. In order to carry out these responsibilities relative to the nuclear power industry, the Environmental Protection Agency has performed field studies at nuclear power stations and related facilities. These field studies have required the development of means for identifying and quantifying radionuclides as well as the methodology for evaluating reactor plant discharge pathways and environmental transport.

Electrical generation utilizing light-water-cooled nuclear power reactors is experiencing rapid growth in the United States. The growth of nuclear energy has been managed so that environmental contamination is minimal at the present time. The Environmental Protection Agency has engaged in studies at routinely operating nuclear power stations to provide an understanding of the radionuclides in reactor effluents, their subsequent fate in the environment, and the real or potential population exposures.

A previous study at the Yankee Rowe reactor (185 MWe) provided an initial base for evaluating the environmental effects of operating pressurized water reactors. This particular field study was performed at the Haddam Neck (formerly called Connecticut Yankee) nuclear power station, a 590 MWe pressurized water reactor. Results from this study have allowed the evaluation of the operational and environmental effects of larger pressurized water reactors, and will provide a better basis on which to evaluate even larger reactors not yet operating.

Comments on this report would be appreciated. These should be sent to the Director, Technology Assessment Division of the Office of Radiation Programs, Environmental Protection Agency, 401 M Street, S.W., Washington, D.C. 20460.

W. D. Rowe

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1. INTRODUCTION

1.1 Need for Study

Determining the potential radiation exposure beyond the station boundary due to routine operation is one of several important aspects in evaluating the impact of a nuclear power station on the environment and the public. This determination requires detailed knowledge of radioactive discharges at the station and environmental pathways leading to radiation exposure. A useful approach is to quantify this knowledge in terms of a "model" station in a "model" environment. The calculational models must then be confirmed or appropriately modified by observation and measurement at each station under consideration. Several aspects of these models are in their early stages and require the collection of additional radioactivity data at nuclear power stations. In some cases, even the procedures for obtaining these data must be developed and tested.

The Office of Radiation Programs, U. S. Environmental Protection Agency (EPA), has, for these reasons, undertaken a program of studies at commercially operated nuclear power stations. The U.S. Atomic Energy Commission (AEC), state health or environmental protection agencies, and station operators are cooperating in this program. This report describes the third of four projects—two at boiling water reactors (BWR's) and two at pressurized water reactors (PWR's). Results of the first two projects, at the Dresden BWR and the Yankee-Rowe PWR, have been published.(1,2)

The four stations were selected for study so as to provide generally applicable information. Because the program was begun during the initial expansion in nuclear power production when only a few relatively small stations were at full operation in the U. S., care must be taken in applying observations to newer stations that are larger and inevitably different in design and operation. To make the study at this third station generically applicable, results have been related to the calculational models presented by the AEC and EPA, (3,4) although these became available only after the study was completed. The pertinence of observations at this station for estimating exposures at other stations can also be evaluated by comparing the amounts of discharged radionuclides and the pattern of these discharges. Gross radioactivity values are given for this purpose in Section 1.2; annual discharges of individual radionuclides are now routinely reported by station operators and compiled by the AEC;(5) and the separation of data for effluent sources within the station, as shown in Section 8.1, may well become available in future compilations.

Guidance for evaluating environmental radiation exposure by emphasizing the observation of critical radionuclides, pathways, and exposed populations in the environment has been available for some time.(6) This approach concentrates efforts on the few most important ("critical") causes of exposure in the presence of many potential ones. Models for computing radionuclide transfers—for example, from water to fish, stack to vegetation, and stack to cows' milk for ^{131}I—have been utilized in the two earlier reports(1,2) and are described fully in the AEC and EPA models.(3,4) The two agencies have recently published environmental monitoring guides,(7,8) and appreciable information concerning environmental transfers at nuclear facilities, beyond that cited in the earlier reports, has been presented in the past few years.(9-15) At least two other detailed studies at commercial nuclear power stations in the U. S. are available, one of radionuclides in the aquatic ecosystem(16) and the other of terrestrial radiation exposure.(17)

The methods for collecting and analyzing samples have been described in this series of reports with some care because test measurements at and beyond the station are important in validating computed exposures. The problems encountered in sampling and analysis at the station are very different from those in the environment. Samples taken at the station usually show numerous radionuclides, including some that decay rapidly or are not readily detectable by the preferred method of gamma-ray spectrometry. Samples from the environment, on the other hand, usually contain only a few radionuclides at very low concentrations, distinguishable from background radiation only with difficulty, if at all. Analytical efforts for in-plant samples, therefore, must be focused on effectively resolving complex mixtures, while environmental monitoring requires informed sample selection and ultra-low-level analyses.

1.2 The Station

The study was undertaken at the Haddam Neck (also called Connecticut Yankee) Nuclear Power Station, a PWR station built by the Westinghouse Electric Corp. for the Connecticut Yankee Atomic Power Co. The station began operating in 1967 and reached its present maximum power level of 1825 megawatts thermal (MWt) in 1969; the corresponding gross electrical output is approximately 590 megawatts (MWe). It had produced 15 terawatt-hours (TW-hr) of electricity at the end of 1971—more than any other commercial nuclear power station in the U. S. Operation of the station is described in several publications.(18-22)

At the time the study began in June 1970, the reactor had been partially refueled once, in April 1970. Most fuel elements inserted in 1970 consisted of uranium oxide (UO_2) pellets enriched to 3.67 percent in ^{235}U, and clad in stainless steel. The ^{235}U enrichment was lower in fuel elements loaded earlier, and higher in the subsequent loading on April, 1971. A few Zircaloy-clad test elements were also in the reactor during the study period.

The station is located in a shallow valley on the east bank of the Connecticut River. It is in the town of Haddam Neck, Connecticut, 35 km south-southwest of Hartford and 26 km north of Long Island Sound.

The study was undertaken at Haddam Neck because it was one of the two larger PWR stations in the U. S. that had been in operation for more than a year in 1970. For comparison, the commercial PWR stations that had been operated a full year in 1972 are listed below with their radioactive discharges in curies (Ci) during that year:(8)

Station	Year of initial operation	Rated power, MWe	1972 power generation, TW(t)-hr	Liquid waste, Ci: Gross beta	Liquid waste, Ci: ^3H	Airborne waste, Ci: Gases	Airborne waste, Ci: Particles & iodine
Yankee Rowe	1960	175	2.4	0.02	803	18	0.0008
Indian Pt. I	1963	265	2.7	25.4	574	543	0.93
Palisades	1972	710	5.9	6.8	208	505	0.0097
R. E. Ginna	1969	420	7.7	0.38	119	11,800	0.035
San Onofre I	1967	430	8.5	13.2	3,480	19,100	0.0005
Point Beach I	1970	497	10.0	1.53	563	2,810	0.030
Haddam Neck	1967	590	13.8	4.8	5,890	645	0.018
H. B. Robinson	1970	663	15.0	0.83	405	257	0.027

All stations were built by Westinghouse except Indian Point I (built by Babcock and Wilcox) and Palisades (built by Combustion Engineering). A notable difference in those that began operating after 1967 is the use of fuel clad in Zircaloy instead of stainless steel. The gross activity at Haddam Neck in both liquid and airborne waste is shown by the above table to have been median to values at other stations. The relatively high amounts of ^3H in liquid waste at Haddam Neck and other older stations are attributed to fission-produced ^3H leaking through stainless steel cladding.

1.3 The Study

The study was undertaken in seven field trips to the station and its environs between June, 1970 and April, 1971. The trips were scheduled to observe radionuclide concentrations throughout the station operating cycle at various seasons. Such observations, under ordinary circumstances, were considered to approximate average or total radionuclide values for sources and pathways sufficient for the generic purpose of the study. The computed averages or totals from this study are compared, when possible, with values obtained by the station operator, often by much more frequent analyses, to evaluate the applicability of the measurements during the field trips. The field trips were not intended to be inspections of operating practices at the station.

The study was performed by the Radiochemistry and Nuclear Engineering Facility at the EPA National Environmental Research Center, Cincinnati, with the support of the Technology Assessment Division, ORP-EPA, and other EPA laboratories. Cooperating in these studies were the persons listed in Appendix A from the Connecticut Department of Environmental Protection, the operating utility, and AEC. The utility's contractor for aquatic studies, the Essex Marine Laboratory, directed by Professor Daniel Merriman, was particularly helpful in collecting aquatic samples and giving guidance in their field.

The study had been planned on the basis of results obtained at the similar but smaller PWR station at Yankee-Rowe.(2) In addition, the following information provided guidance: publications describing the Haddam Neck station,(18-22) monthly station operating reports, reports by the operator's contractor for environmental surveillance,(23) and the State's environmental surveillance reports.(24)

This information suggested that:

(1) several sources at the station would emit gaseous and liquid effluents of comparable dosimetric import;
(2) critical radiation exposure pathways would include consumption of fish caught near the outfall, direct radiation from waste storage tanks, and external radiation from effluent gases;
(3) bottom sediment and aquatic vegetation near the liquid waste outfall would be among the few environmental media to contain readily detectable radionuclides from the station;
(4) radioiodine might be at detectable levels in the thyroid of cattle grazing near the station.

The measurement program accordingly emphasized these aspects of the station and its environment.

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2. RADIONUCLIDES IN WATER ON SITE

2.1 Water Systems and Samples

2.1.1 *General.* The power-producing systems at the Haddam Neck plant are typical of PWR's. Water flows in the reactor coolant, secondary coolant and chemical and volume control systems shown in Figure 2.1. Other water systems on site include boron recovery and waste disposal, spent fuel pit cooling, safety injection, component coolant, circulating water, service water, and sanitary water.

2.1.2 *Reactor coolant system.*(1) During routine operation, reactor (primary) system water under a pressure of 2,000 psig is heated in the reactor by the fission process in nuclear fuel. The water is pumped to four steam generators in parallel at a total flow rate of 4.6×10^7 kg/h (1.0×10^8 lb/h). Approximately 1.6×10^5 kg (3.6×10^5 lb) of water are in a system volume of 2.5×10^5 liters (8,782 ft^3). The water temperature is 318° C at the reactor outlet and 291° C at the inlet at full power.

Reactor coolant water contains lithium hydroxide and, during most of the fuel cycle, boron in the form of boric acid. The boron is added to supplement the control rods for maintaining criticality. It is gradually decreased from 800 mg/kg immediately after refueling to 0 mg/kg after 8 months of operation during a cycle of 10–11 months of operation and 1–2 months for refueling (see Figures 2.2 and 2.3). The concentration of lithium is approximately 1 mg/kg throughout the operating cycle; additional lithium hydroxide is added when the lithium concentration goes below 0.5 mg/kg. The lithium hydroxide maintains an acidity for corrosion control at approximately pH 6 (measured at 25° C) in the presence of boric acid and at pH 10 in its absence. Monthly averages of these values, reported by the station operator, are summarized in Appendix B.1.

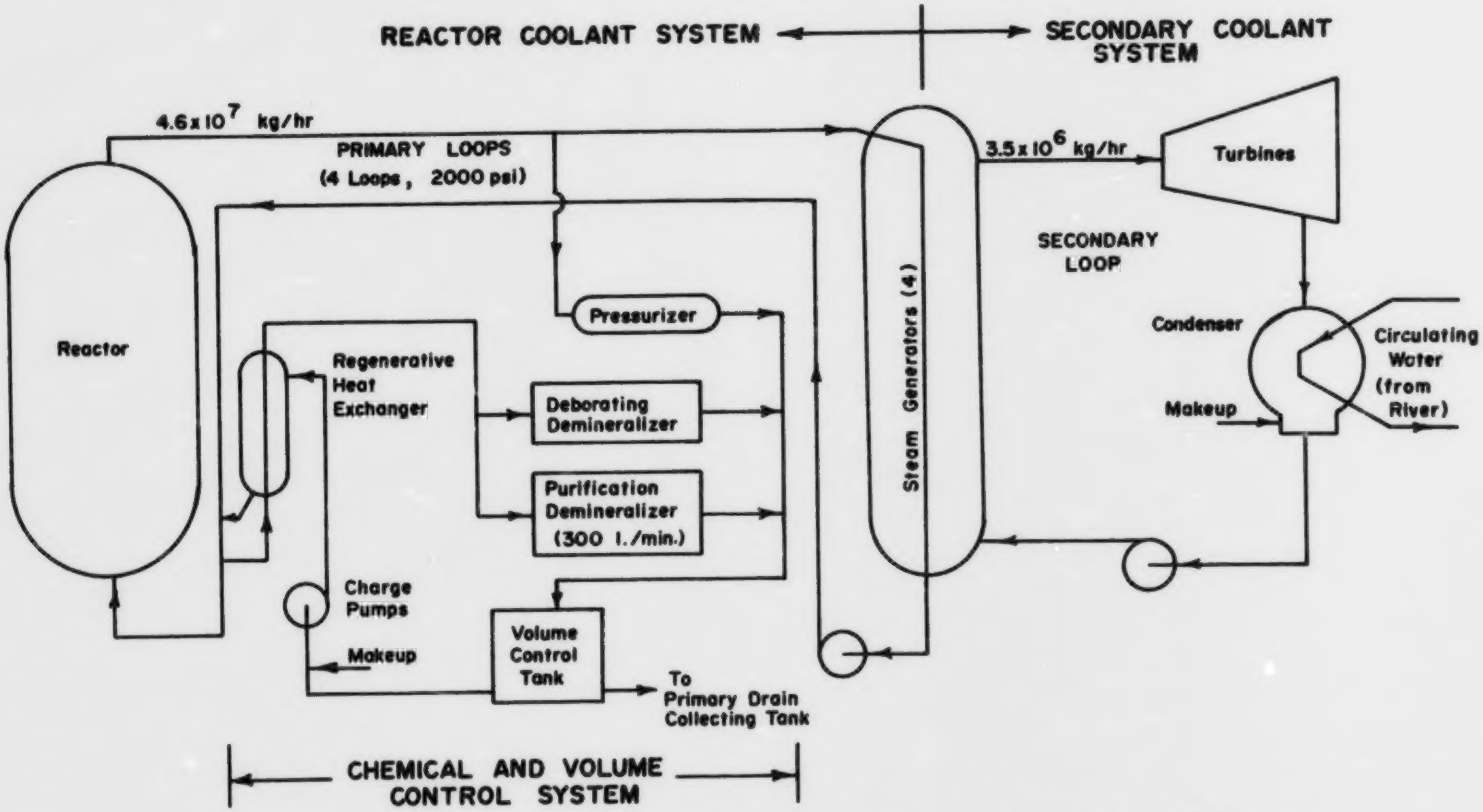
The water is under nitrogen gas to provide an inert atmosphere. Hydrogen gas is added to keep the amount of oxygen from the radiation-induced decomposition of water below 0.1 mg/kg. The concentration of hydrogen is usually 30–35 cc/kg; that of nitrogen is approximately 1 cc/kg at standard temperature and pressure.

2.1.3 *Chemical and volume control system.*(1) The chemical and volume control system (CVCS) is used to adjust the pressure, volume, purity and chemical content of the reactor coolant system (see Figure 2.1). Makeup water is obtained from two shallow wells on site, demineralized by passing it through mixed-bed ion-exchange resin, stored in a 570,000-liter (150,000-gal) tank, and heated before it is added to the reactor coolant. One stream of the reactor coolant flows continuously at the rate of 300 kg/min through the purification demineralizer—a mixed-bed ion-exchange-resin column with 25-micron filter—to maintain the purity of the water in order to prevent deposition on heat-transfer surfaces and in flow channels.

For reactor shutdown, water at high boron concentration (several thousand mg/kg) is added through the system; at startup, the boron concentration is lowered to the appropriate operating values by feed-and-bleed replacement with water. To remove the last amounts of boron (below 30 mg/kg) near the end of the operating cycle, reactor coolant water is circulated through the deborating demineralizer.

2.1.4 *Secondary coolant system.*(1) The secondary coolant system contains 2.6×10^5 kg of water. The water is converted to steam at a nominal pressure of 690 psia and temperature of 261° C in four steam generators. Each steam generator contains approximately 3.0×10^4 kg (8×10^3 gal) of water.(1) The steam operates a high pressure turbine and, after passage through four moisture separator reheaters, two low-pressure turbines. The used steam is condensed at an absolute pressure of 38 mm (1.5 in) Hg in two condensers that have a hot-well capacity of 82,000 liters (21,600 gal) each.

The condensed water, together with steam and water from the moisture separators, is pumped through feedwater heaters (omitted from Figure 2.1 for the sake of simplicity) back to the steam generators at a feedwater temperature of 222° C. Makeup water is pumped from the wells on site, demineralized, stored in a 380,000-liter (100,000 gal) tank, and delivered to the condenser hot-wells as needed.


REACTOR COOLANT SYSTEM
SECONDARY COOLANT SYSTEM
4.6 x 10^7 kg/hr
PRIMARY LOOPS
(4 Loops, 2000 psi)
Reactor
Regenerative Heat Exchanger
Pressurizer
Deborating Demineralizer
Purification Demineralizer (300 l./min.)
Charge Pumps
Makeup
Volume Control Tank
To Primary Drain Collecting Tank
CHEMICAL AND VOLUME CONTROL SYSTEM
Steam Generators (4)
3.5 x 10^6 kg/hr
Turbines
SECONDARY LOOP
Condenser
Makeup
Circulating Water (from River)

POWER - 1825 MWt
WATER
REACTOR SYSTEM - 1.6×10^5 kg
SECONDARY SYSTEM - 2.6×10^5 kg

Figure 2.1 Coolant Flow Schematic

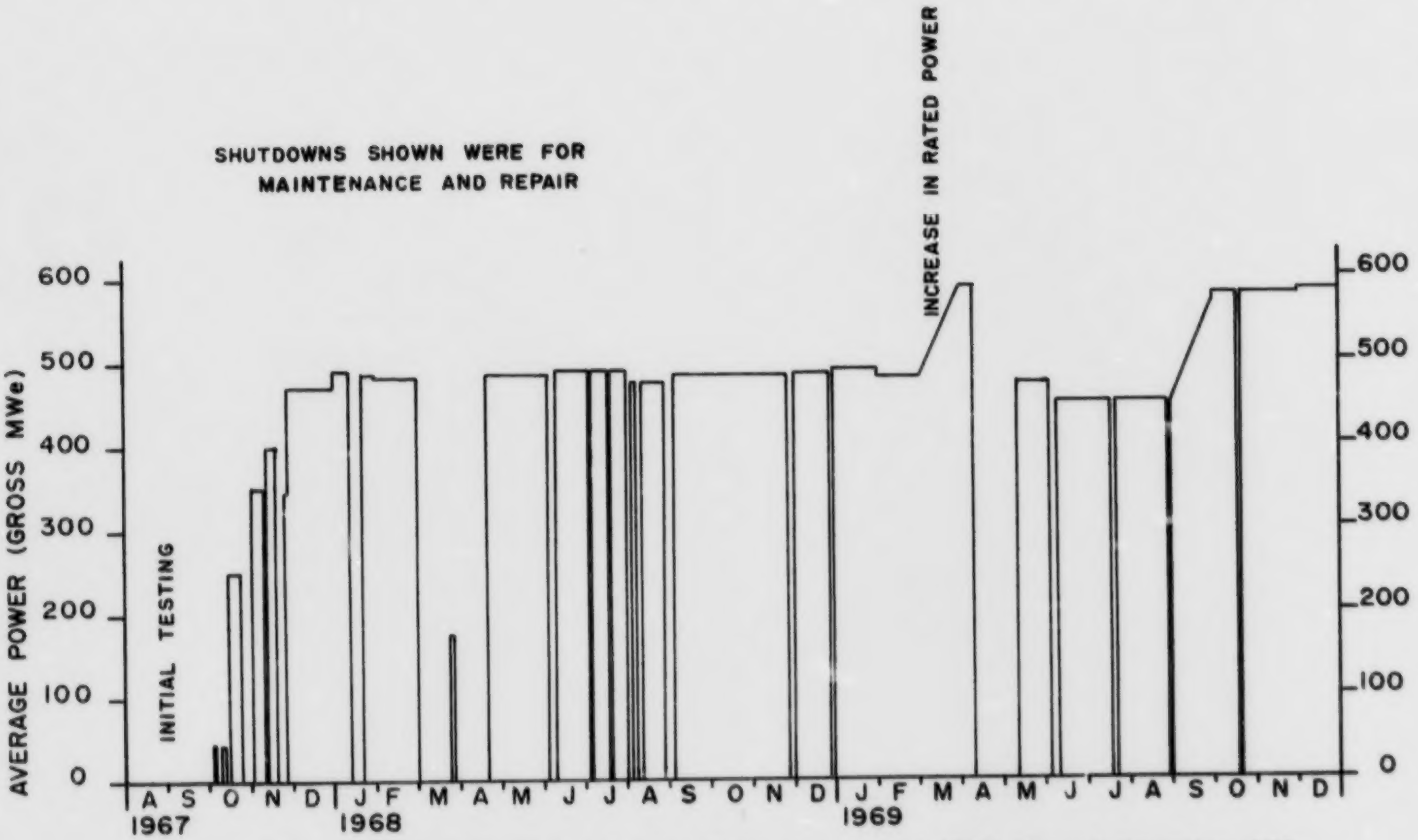


Figure 2.2 Haddam Neck Electrical Production, August 1967 Through December 1969
(From Monthly Operation Reports)

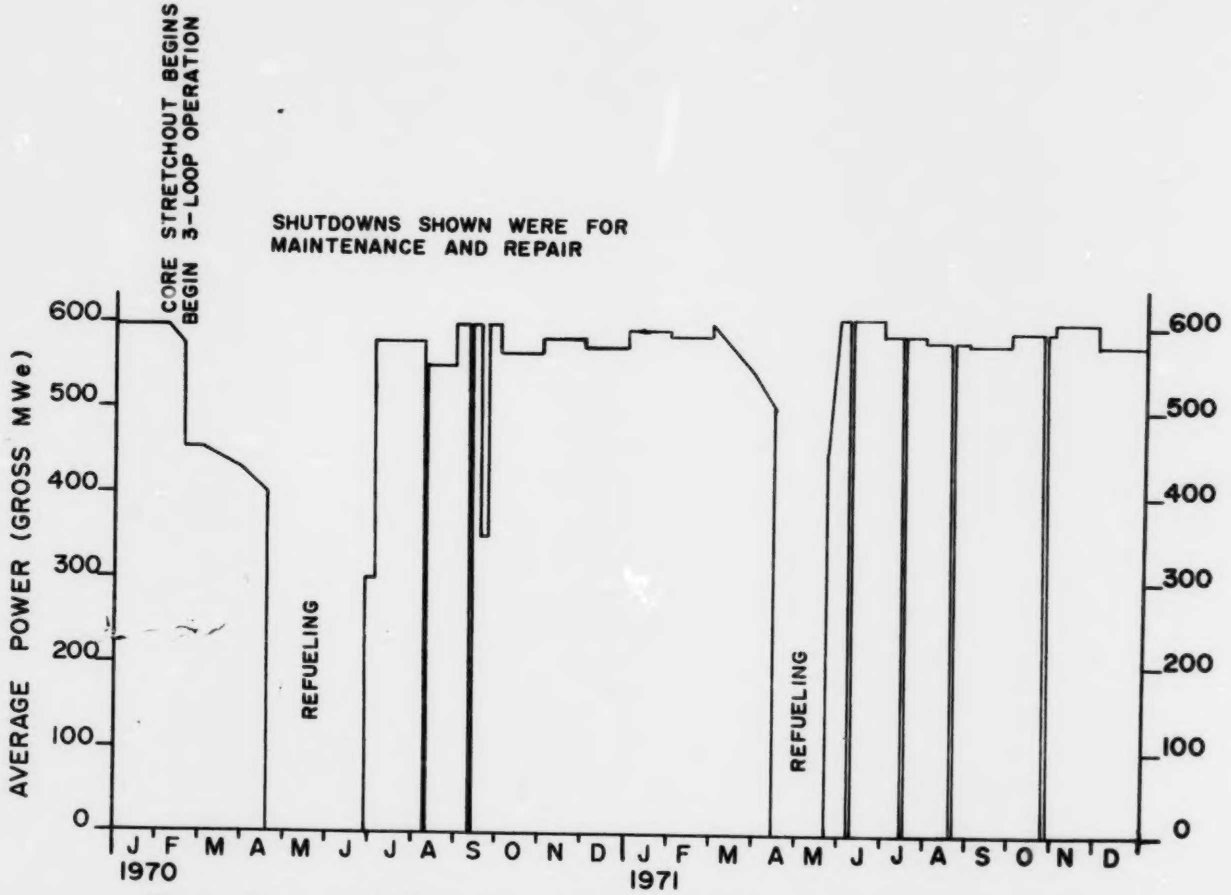


Figure 2.3 Haddam Neck Electrical Production, January 1970 Through December 1971 (From Monthly Operation Reports)

2.1.5 *Paths of radionuclides from reactor, CVCS, and secondary systems.*(1,2) The radionuclides in the reactor coolant water are fission products and activation products. The fission products in the water are formed within the uranium oxide fuel and enter the water through small imperfections in the stainless steel cladding of the fuel elements. Other possible sources of fission products—apparently minor—are fuel that contaminates the surface of new fuel elements ("tramp uranium") and fuel that passes into reactor coolant water from failed fuel elements. The activation products in reactor coolant water are formed by neutron irradiation of the water and its contents (including gases and dissolved or suspended solids) and of materials in contact with the coolant (container and structural surfaces, fuel and control rod cladding) that subsequently corrode or erode.

The radionuclides in the reactor coolant water circulate and decay within the system, deposit as "crud" (which may later recirculate), are retained by the purification and deborating demineralizers (which are periodically replaced and shipped off-site as solid waste), or leave the system with gases and liquids. Paths from the system to the environment are shown schematically in Figure 2.4.

During routine operation, water and associated gases leave the reactor coolant system through leaks, intentional discharge from the volume control tank, and sample collection. At the time of the study, the total water leakage rate (see Appendix B.1) averaged 2,200 kg/day. At this rate, intentional discharge ("shim bleed") was not required for routinely reducing the boron concentration.

Of the total leakage, 75 to 150 kg/day (20 to 40 gal/day) were reported to leak into the secondary system through steam generator tubes.(3,4) Various leaks into the vapor containment were estimated(3) to total 110 kg/day (40 gal/day at specific gravity of 0.7), of which approximately 35 percent initially would turn to steam. After several weeks, the building atmosphere will be saturated with water vapor and an amount of water equal to that leaking will collect in the containment building sump; the measured rate of water transfer to the aerated liquid waste system averaged 280 kg/day.(1) Leakage into the primary auxiliary building has been estimated to occur at the rate of 75 kg/day (20 gal/ day),(3) of which 2.6 kg/day is steam,(4) and the remainder is water collected as aerated liquid waste. The largest amount of leaking water—approximately 1900 kg/day—is collected in reactor coolant drains in the containment and primary auxiliary buildings and passes to the boron recovery system.

The following amount of reactor water was discharged in 1971 (see Appendix B.3):

boron recovery ("hydrogenated" liquid waste) system	**1.8×10^6 kg/year**
"aerated" liquid waste system	**3.0×10^6**
total	**4.8×10^6 kg/year**

The leakage rate of 1,900 kg/day contributes 0.6×10^6 kg/year to the boron recovery system in 330 days of operation. Most of the water enters the boron recovery system during reactor system shut-down and start-up, refueling, and boron removal toward the end of reactor operation. The amount of water from the containment building sump (9×10^4 kg/yr) is only a small fraction of the aerated liquid waste.

Radionuclides in the leaking water are expected to be at the concentrations observed in samples of reactor coolant water, except for the effects of steam flashing (i.e., higher concentrations of nonvolatile radionuclides and lower values for volatile radionuclides). Volatile radionuclides accumulate in the containment building until that building is vented. This occurs at least once and possibly a few times each year (see Section 3.1). In the primary auxiliary building, ventilating air, which contains volatile radionuclides, is discharged continuously. Radionuclides in the shim bleed are at lower concentrations than in reactor water samples to the extent that these radionuclides are removed by the purification demineralizer (see Figure 2.1).

Radionuclides enter the secondary coolant system through steam-generator leaks. Average leakage rates were reported to be as follows in 1971:(5)

February	**110 kg/day**
September	**55**
October	**150**
November	**150**
December	**150**

Occasionally, the leakage rate increases rapidly until the faulty tubes are plugged. In February, 1970, the reactor-to-secondary leakage reached 6,400 kg/day, whereupon the loop at fault was isolated, and the leaking tubes were plugged during refueling in April.

Water leaves the secondary system through steam generator blowdown, discharge of moisture-saturated noncondensable gases and system leakage. The steam generators are blown down continuously during the day at the rate of 2,300 kg/day. The additional nightly blowdown is 21,000 kg/day during 8 hours if inleakage from the reactor system is appreciable, or 7,000 kg/day during 4 hours for negligible inleakage. An average blowdown rate of 19,000 kg/day has been reported.(5) It was estimated that 35 percent of this amount is

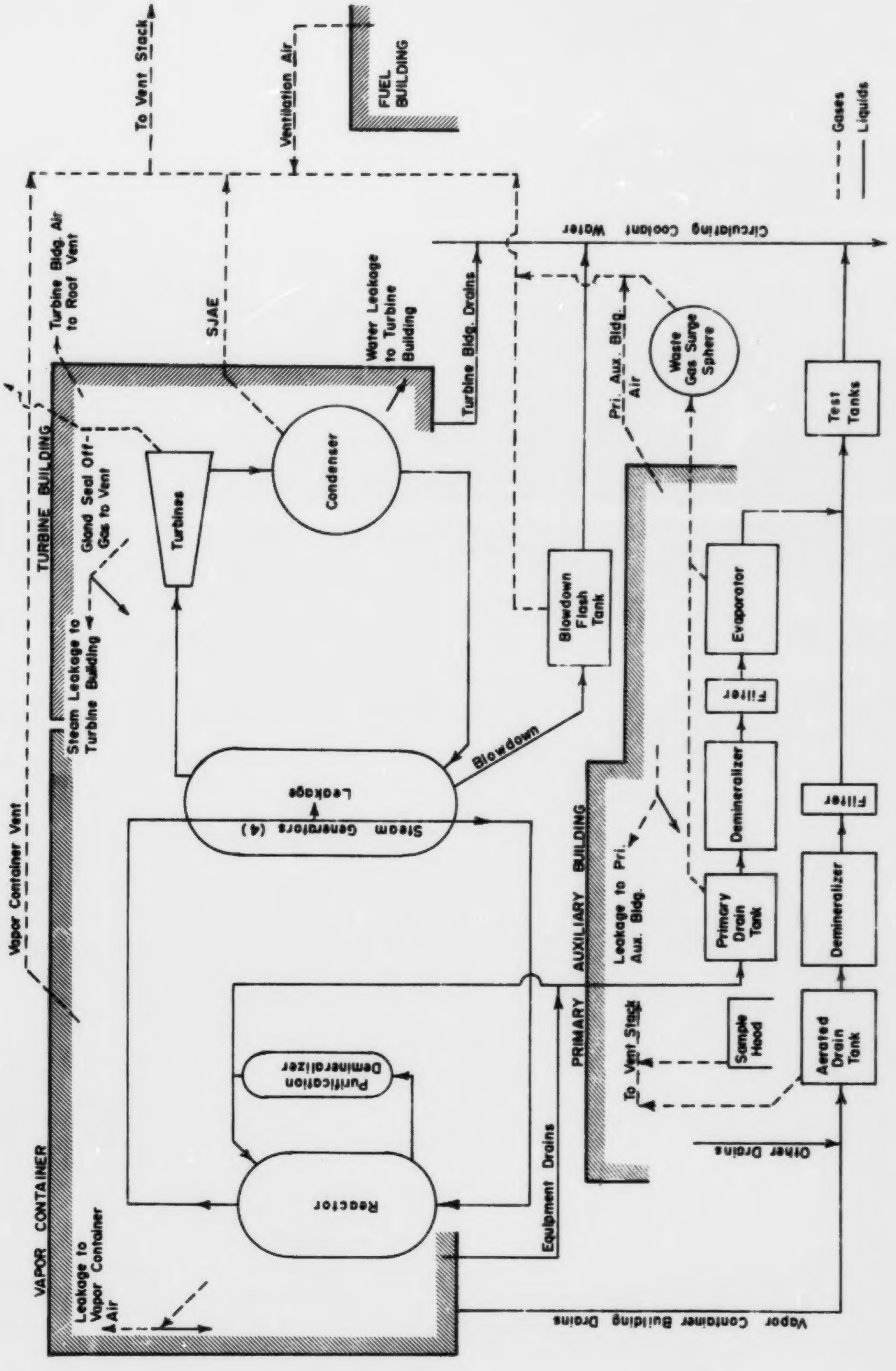

To Vent Stack
Ventilation Air
FUEL BUILDING
Turbine Bldg. Air to Roof Vent
SJAE
Water Leakage to Turbine Building
Turbine Bldg. Drains
Circulating Coolant Water
TURBINE BUILDING
Gland Seal Off-Gas to Vent
Turbines
Condenser
Pri. Aux. Bldg. Air
Waste Gas Surge Sphere
Test Tanks
Blowdown Flash Tank
Evaporator
Filter
Demineralizer
Steam Leakage to Turbine Building
Blowdown
Vapor Container Vent
Steam Generators (4)
Leakage
PRIMARY AUXILIARY BUILDING
Leakage to Pri. Aux. Bldg.
Primary Drain Tank
Filter
Demineralizer
Purification Demineralizer
To Vent Stack
Sample Hood
Aerated Drain Tank
VAPOR CONTAINER
Reactor
Equipment Drains
Other Drains
Leakage to Vapor Container Air
Vapor Container Building Drains
Gases
Liquids

Figure 2.4 Effluent Release Pathways

vented to the atmosphere from the flash tank as steam;(4) the remainder condenses and is discharged to the coolant canal. All but the volatile radionuclides are expected to remain in the water.

Noncondensable gases are removed continuously from the secondary system by the air ejectors at the main condensers and through the turbine gland seals. The gases include volatile radioisotopes of krypton and xenon. The discharge rate of moisture is approximately 30 kg/day at the main condenser (Section 3.3.5) and may be somewhat more at the gland seal condenser (Section 3.3.7).

Leakage from the secondary system plus discharge with noncondensable gases was approximately 38,000 kg/day at the time of the study.(1) The rate of steam leakage, if it is one-half of the rate for a "model" 3,500 MWt plant,(4) would be 9,300 kg/day. Of this, 3,300 kg/day (35 percent) would remain as steam and 6,000 kg/day would condense.(4) The steam is discharged with turbine-building air, while the water is collected in turbine-building drains together with water leaking from the secondary system at the rate of 28,700 kg/day, (i.e., 38,000 minus 9,300 kg/day). The water is discharged directly to the coolant canal. Both steam and water are expected to contain mostly tritium and volatile radionuclides.

The annual discharge from the secondary coolant system for 330 days of operation, based on the above values, is:

blowdown water	**4.1 x 10^6 kg/year**
blowdown steam	**2.2 x 10^6**
leakage water	**11.5 x 10^6**
leakage steam	**1.1 x 10^6**
total	**18.9 x 10^6 kg/year**

The computed annual discharge rate of water plus steam from the reactor and secondary coolant system is thus 4.8 x 10^6 + 18.9 x 10^6 = 23.7 x 10^6 kg/yr. This sum is almost the same as the 1970 total reported by the station operator, but is less than the 1971 discharge (see Section 4.1.1). Considerable fluctuations may occur from year to year due to changes in water use and leak rates.

2.1.6 *Other liquids on site.*(1) Several ancillary water systems exist at the station, but only the first three of the following are believed to result in radioactive discharges:

(1) Boron recovery and waste disposal system. The system for liquids is described in Section 4.1.1, and for gases, in Section 3.1.1.

(2) Spent-fuel pit. Fuel pit water consists of approximately 1.3 x 10^6 kg (3.5 x 10^5 gal). Demineralized water from wells on site is pumped through the fuel pit in which used fuel elements are stored. During reactor operation, the water is circulated continuously through a mixed-bed ion-exchange demineralizer at 4 kg/s. This water is not discharged during reactor operation.

(3) Refueling water. The reactor cavity and refueling canal connecting the reactor vessel with the fuel pit are flooded during refueling with 8.9 x 10^5 kg (2.35 x 10^5 gal) of borated (1820 mg/kg) water from the 9.8 x 10^5-kg (2.6 x 10^5-gal) refueling water storage tank. The top of the reactor vessel is opened and fuel elements are moved from the reactor vessel to the fuel pit while submerged in the water. After storage for decay of short-lived fission products, the fuel is transferred to casks and shipped off-site. The water from the refueling cavity is circulated through the purification demineralizer during refueling. Water leaking or pumped from this system is transferred to the boron recovery and waste treatment system. After refueling, the water in the reactor cavity is returned to the refueling water storage tank via the residual heat removal system.

At the beginning of the long, cold shutdown for maintenance in April 1969, soluble and insoluble ^{58}Co were released to the refueling water,(6) causing high surface exposure rates on many auxiliary plant components. A possible source of the ^{58}Co is the large amount of nickel-bearing alloys in contact with the water. Provision was made to pass the water through the purification demineralizer during shutdown. During refueling in 1970 and 1971, increased ^{58}Co concentrations in the water were reduced satisfactorily by the purification system.

(4) Safety injection system. During reactor operation, the safety injection and core deluge systems would use the water in the refueling water storage tank if these systems were needed. The borated water can be rapidly pumped into the reactor core in case of a major loss-of-cooling accident.

(5) Component cooling system. Water for the component cooling system consists of 30,000 kg (8,000 gal). Potassium chromate (175–225 mg/kg) is added as a corrosion inhibitor. The system is an intermediate cooling system to transfer heat from components containing

reactor coolant to the service water cooling system. This water is not normally discharged.

(6) Circulating water system. Circulating cooling water is pumped from the Connecticut River through the main condenser by 4 pumps at the rate of 1.4 x 10^6 kg/min (327,000 gal/min). It is returned through a canal to the river, carrying with it the heat extracted from the steam. The maximum temperature increase in the circulating cooling water is 12.4° C (22.3° F).

(7) Service water. Connecticut River water is pumped into the service water system to cool several systems at the station, clean the traveling screens in the circulating coolant water intake, and inject hypochlorite periodically into the circulating coolant water for eliminating bacterial slime. Service water cools most station auxiliary equipment, including the component cooling system, spent fuel pit water, and miscellaneous oil and air coolers. Three of the four 23,000 kg/min (6,000 gal/min) pumps provide the required flow from the circulating coolant water intake. The service water is discharged into effluent circulating coolant water.

(8) Sanitary waste. This water is discharged into two 11,000 kg (3,000 gal) septic tanks, one at the plant and the other at the information center.

2.1.7 *Samples.* To identify potential radioactive effluents, liquids at the Haddam Neck Nuclear Power Plant were sampled within the plant where radionuclides were at much higher concentrations and, therefore, more easily detected than at the point of release. The following water samples were provided in plastic bottles by station personnel:

1. reactor coolant, 2 liters, collected July 24, 1970 at 0900;
2. reactor coolant, 1.5 liters, collected Nov. 20, 1970 at 0830;
3. reactor coolant, 1 liter, collected March 16, 1971 at 1000;
4. steam generator blowdown, 3.5 liters (pH 6.8), collected July 23, 1970 at 1500 hours;
5. composite steam generator blowdown, 3.5 liters, collected Sept. 15, 1970 at 1000;
6. steam generator blowdown, 3.5 liters, collected Nov. 20, 1970 at 0830;
7. steam generator blowdown, 3.5 liters, collected Mar. 15, 1971 at 1000;
8. steam generator blowdown, 3.5 liters, collected Mar. 16, 1971 at 1000;
9. steam generator blowdown, 3.5 liters, collected April 14, 1971 at 1945;
10. refueling cavity water, 1 liter, collected May 7, 1971, at 1115.

One liter each of samples #1, 2, and 6, and 500 ml of sample #3 were acidified with conc. HNO_3 (10% v/v) to minimize deposition of radionuclides on the walls of the bottle. The unacidified portion was reserved for radioiodine, tritium, and ^{14}C analyses.

2.2 Analysis

2.2.1 *General.* Aliquots of all samples were counted for gross alpha and beta activity, examined with gamma-ray spectrometers and analyzed radiochemically. Analyses were performed for high-yield fission products and common activation products in reactor water. Because radioactive decay between sampling and analysis was usually between 1 and 2 days, radionuclides with half-lives less than 6 hours, and in some cases, 24 hours, could not be measured. Aliquot volumes ranged from 1 to 200 ml.

Radionuclide concentrations were computed from count rates obtained with detectors calibrated with radioactivity standards. Values were corrected for radioactive decay and are given as concentrations at sampling time. Half-lives and branching ratios are from recent publications.(7-10) The concentration of radioactive progeny such as ^{133}Xe, ^{133m}Xe, and ^{135}Xe was corrected for ingrowth in the sample between collection and analysis.

The difficulty of retaining radionuclides in solution reported earlier(11) was also observed during this study, in that radionuclides remained on empty plastic sample containers when the liquid samples were poured out after contact periods of days to weeks. Even with acidification, losses of 10–50 percent were observed for radionuclides such as ^{51}Cr, ^{54}Mn, ^{58}Co, ^{60}Co, and ^{59}Fe. The following techniques were applied to prevent underestimating the radionuclide content of liquid samples:

(1) Cutting the empty sample bottle into small pieces, placing it in a container of known counting efficiency, and measuring gamma-ray emitters.

(2) Collecting the liquid sample on a dry sponge in a container to saturate the sponge with the liquid at a volume calibrated for the counting efficiency of gamma-ray emitters.

(3) Passing samples of low ionic content immediately through cation- and anion-exchange membrane filters* to collect particulate and ionic radionuclides on the filters(12) for analysis by a gamma-ray spectrometer. The filtrate was also analyzed.

(4) Leaching the empty sample bottle with three 25-ml portions of hot aqua regia to collect for analysis those radionuclides that do not emit gamma rays. The completeness of the leaching was checked by assuring that no gamma-ray emitters remained in the bottle. Results obtained for samples treated with procedures (1) or (4) were corrected by including the amounts retained on container walls.

2.2.2 *Radiochemistry.* Radionuclides that emit gamma rays were identified by their characteristic gamma-ray energies with a Ge(Li) detector and 1600-channel spectrometer in aliquots of reactor coolant and liquid wastes. Spectral analyses were obtained at intervals to eliminate interference by shorter-lived radionuclides and to confirm the identity of the measured radionuclides by observing their half lives.

The large number of radionuclides and their concentration differences in coolant water made identification after collection on ion-exchange papers particularly convenient. This technique also identified the ionic form of the radionuclides. Sample #3 (Section 2.1.7) was analyzed in this manner by filtering 35 ml in a suction apparatus through 3 cation- and 2 anion-exchange papers in series. The papers were then separated and transferred individually to containers for spectral analysis. The filtrate was also analyzed. Figures 2.5, 2.6, and 2.7 show the Ge(Li) spectra of each fraction 2 days after collection.

The radionuclides ^{99}Mo-^{99m}Tc, ^{133}I-^{133}Xe, and ^{135}I-^{135}Xe were on the anion paper, and ^{133}Xe and ^{135}Xe were in the filtrate. All other radionuclides were on the top (cation-exchange) filters. These included longer-lived ones at lower concentrations that could only be measured after many of the radionuclides seen in Figure 2.5 had decayed. Chromium-51, although expected to be anionic when dissolved, was retained on the top, cation-exchange filter, possibly because it was in particulate form; some gaseous ^{133}Xe was retained on the same filter, possibly absorbed on the resin.

In reactor coolant samples, gamma rays of energies below 160 keV from relatively short-lived radionuclides were obscured by the radiations from relatively large amounts of ^{133}Xe. The ^{133}Xe was removed by boiling and stirring a 35 ml aliquot of the coolant water with 5 ml conc. HCl. Replicate tests indicated that less than 1 percent of the ^{131}I was volatilized by this process.

A 10-cm x 10-cm NaI(Tl) detector with 200-channel spectrometer was used to analyze samples that contained only a few radionuclides at low levels of radioactivity. The better energy resolution of the Ge(Li) detector was generally unnecessary for these samples, and the higher counting efficiency of NaI(Tl) detectors was advantageous.

A 400-mm^2 Si diode with 400-channel spectrometer was used to identify radionuclides that emit alpha particles in samples that showed detectable gross alpha activity. The alpha-particle energies were determined within ±30 kev. The amount of each alpha-particle-emitting radionuclide was computed from the gross alpha activity and the relative counts at each energy peak.

Radionuclides were separated chemically to confirm gamma-ray spectral identification, measure radionuclides more precisely and at lower concentrations than by instrumental analysis of a mixture, and detect radionuclides that emit only obscure gamma rays or none at all.(13) After chemical separation, the following detectors were used: NaI(Tl) crystal plus spectrometer for photon-emitting radionuclides; low-background end-window Geiger-Mueller (G-M) counter for ^{14}C, ^{32}P, ^{35}S, ^{89}Sr, ^{90}Sr, ^{129}I, ^{147}Pm, and ^{185}W; liquid scintillation detector plus spectrometer for ^{3}H, ^{14}C, and ^{63}Ni; and xenon-filled proportional counter plus spectrometer for ^{55}Fe. Measurements with the G-M detector included observation of the effect of aluminum absorbers on count rates to determine maximum beta-particle energies and thus confirm radionuclide identification.(14)

2.3 Results and Discussion

2.3.1 *Radioactivity in reactor coolant water.* Tritium was by far the most abundant of the measured radionuclides listed in Table 2.1. At second highest concentration was ^{133}Xe; also relatively abundant were

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*Acropor SA-6404 and Acropor SB-6407, Gelman Instrument Company, were found to be satisfactory.

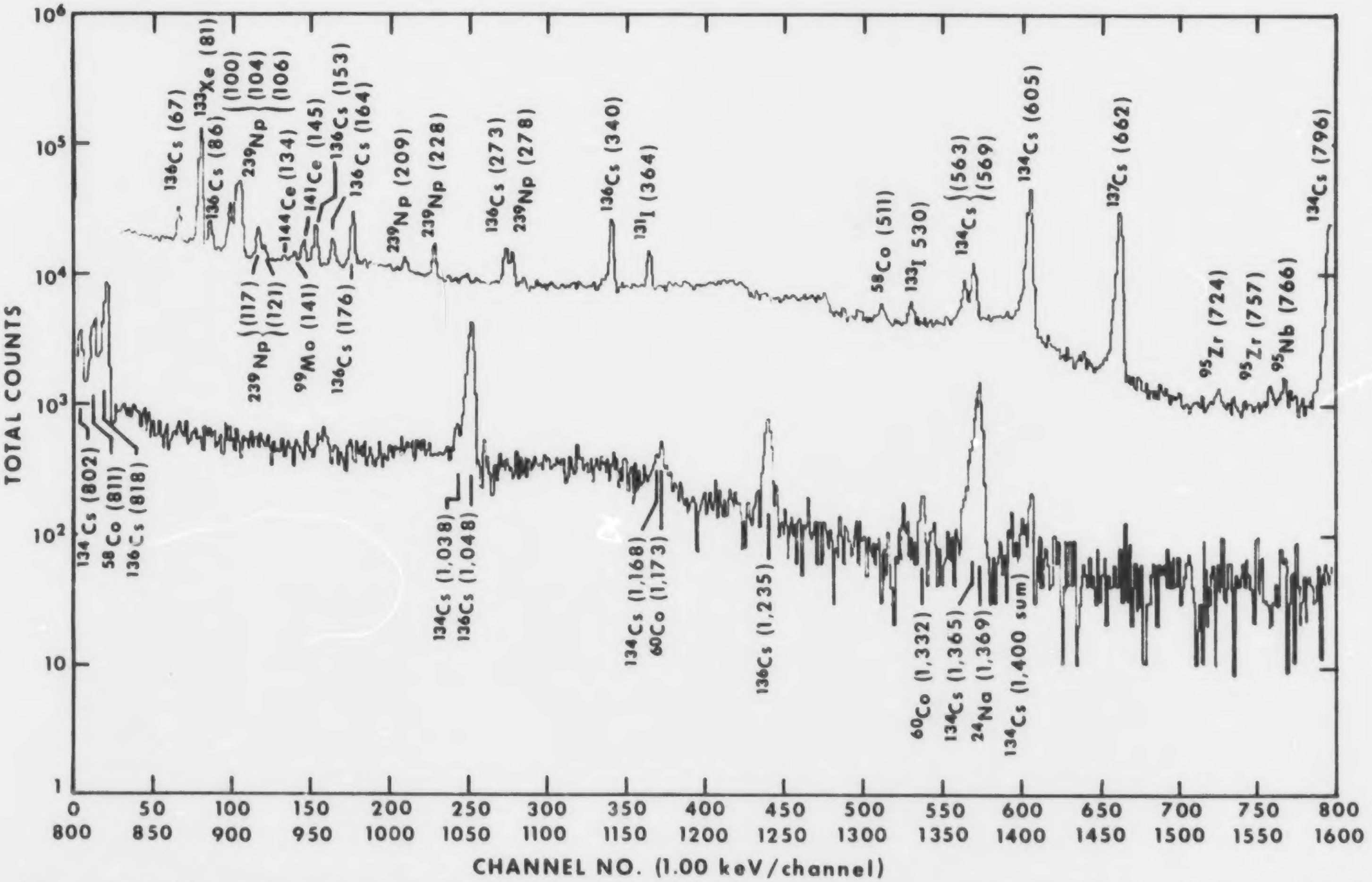

Figure 2.5 Gamma-ray spectrum of radionuclides from reactor coolant retained on cation exchange paper, 0-1600 keV

Detector: Ge(Li), 10.4 cm^2× 11 mm, trapezoidal.

Sample: Cation exchange paper containing activity from 35 ml, collected March 16, 1971 at 1000.

Count: March 18, 1971; 50 minutes (background not subtracted).

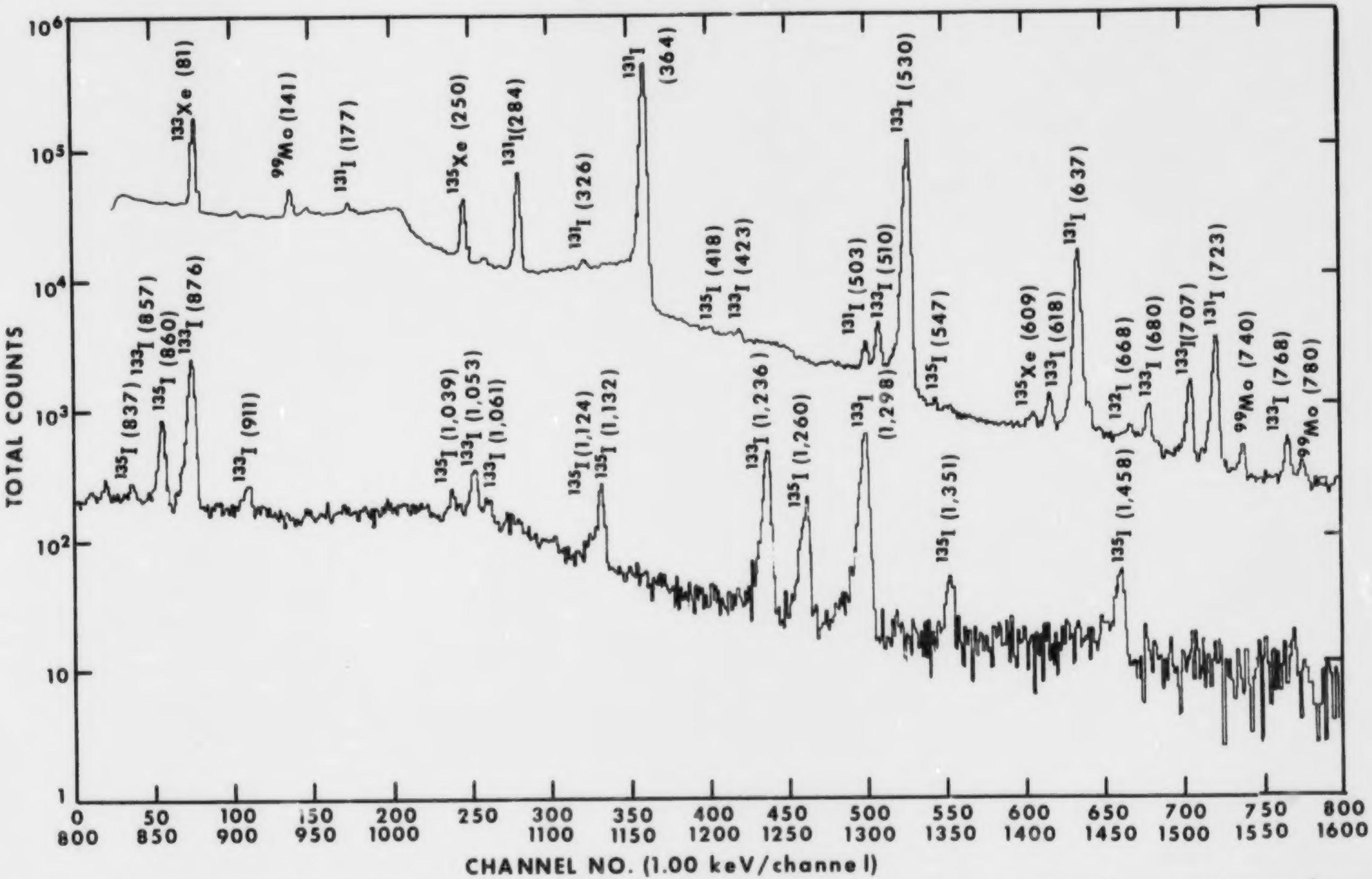

Figure 2.6 Gamma-ray spectrum of radionuclides from reactor coolant retained on anion exchange paper, 0-1600 keV.

Detector: Ge(Li), 10.4 cm^2 × 11 mm, trapezoidal.

Sample: Anion exchange paper containing activity from 35 ml, collected March 16, 1971 at 1000.

Count : March 18,1971: 50 minutes (background not subtracted).

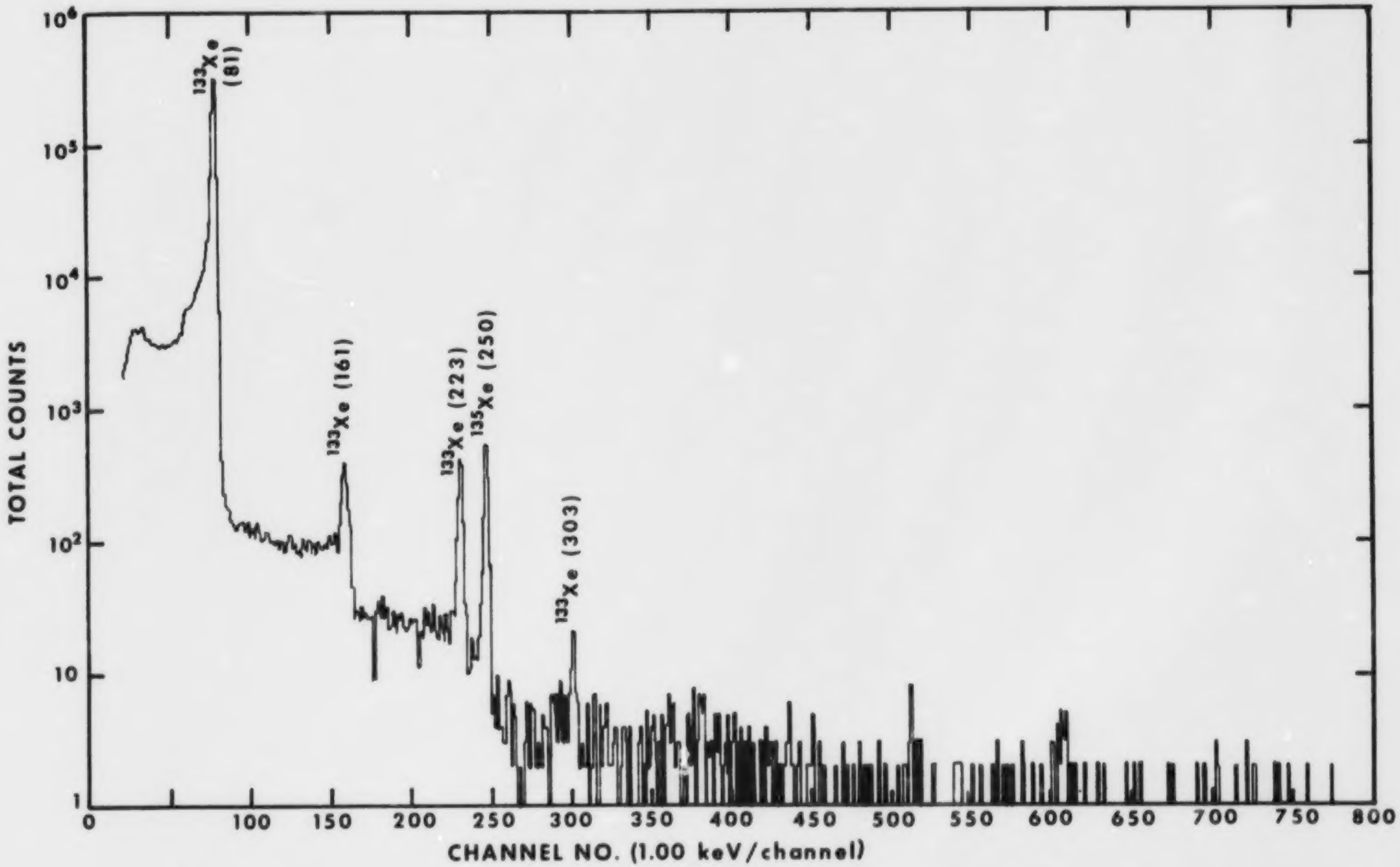

Figure 2.7 Gamma-ray spectrum of radionuclides from reactor coolant not retained on cation or anion exchange papers, 0-1600 keV.

Detector: Ge(Li), 10.4 cm^2 × 11 mm, trapezoidal.
Sample: 35 ml effluent from ion-exchange column, collected March 16, 1971 at 1000.
Count: March 18., 1971; 16.6 minutes (background not subtracted).

^{131}I, ^{133}I, ^{135}I, ^{133m}Xe, and ^{135}Xe. The sum of all other measured radionuclides ranged from 0.002 to 0.01 µCi/ml in the three samples. The sum of all listed radionuclides except ^{3}H and the noble gases was approximately 0.1 µCi/ml. In comparison, the average monthly gross radioactivity excluding ^{3}H (and probably also excluding noble gases) reported by the station during the sampling periods (see below) was between 0.2 and 0.5 µCi/ml. These gross radioactivity measurements may include relatively abundant short-lived radionuclides such as the fission products ^{132}I and ^{134}I and the activation products ^{18}F and ^{56}Mn.

Fission products other than radioiodine and radioxenon were at relatively low concentrations, and several high-yield fission products could not be detected at the limiting sensitivity of approximately 1 x 10^{-6} µCi/ml (see footnote 3 to Table 2.1). Most of the other radionuclides are neutron activation products that have been reported earlier.(11,15) They are formed in water, steel, antimony (in the Sb-Be neutron source), and zirconium (in Zircaloy-2 cladding of 2 fuel rods). The activation products ^{14}C, ^{35}S, and ^{63}Ni were at relatively low concentration, as previously reported for the Yankee-Rowe reactor.(14)

Alpha activity was found in two of the three samples. Isotopic analysis by alpha spectrometer of the highest-level sample of March 16, 1971, showed the following components three years after sample collection:

86-yr ^{238}Pu	9.1 x 10^{-8} µCi/ml
24,400-yr ^{239}Pu and 6,580-yr ^{240}Pu	2.4 x 10^{-8}
17.6-yr ^{244}Cm	0.7 x 10^{-8}
gross alpha activity	12.2 x 10^{-8} µCi/ml

Based on the calculated production of the transuranium elements,(16) the concentration of ^{240}Pu would be slightly greater than that of ^{239}Pu in the unresolved sum of the two isotopes. The decrease of 3.7 x 10^{-7} µCi/ml in three years from the initial gross alpha activity shown in Table 2.1 is attributed to the radioactive decay of 162-d ^{242}Cm. These five radionuclides constitute more than 95 percent of the calculated total alpha activity in the fuel.(16)

Radionuclide concentrations in the three samples were considerably different. This is to be expected because a number of factors change during a fuel cycle; notably, radionuclides accumulate in the fuel, while in the coolant the pH value increases as the boric acid concentration decreases (see Appendix B.1). The monthly average values reported by the Haddam Neck Plant at the sampling periods are:

	July 1970	Nov. 1970	March 1972
Month of core II cycle	lst	5th	9th
Power level, MWe	551	586	571
Boron, mg/kg	571	278	0.6
pH	6.2	6.8	10.0
Gross activity, µCi/ml	0.20	0.26	0.50

Radionuclide concentrations in coolant water are also affected by many other variables, especially the quality of the fuel elements, the rate and effectiveness of coolant-water purification, and the extent of radionuclide accumulation within and loss from the coolant system.

2.3.2 *Tritium in reactor coolant water.* The measured ^{3}H concentrations in Table 2.1 are consistent with the average concentrations of 5.43, 3.68, and 4.52 µCi/ml, respectively, reported for those months by the station operator (see Appendix B.1). The sources of the tritium in coolant water are believed to be, in order of importance: (1) ternary fission in the fuel, (2) ^{10}B (n,2 alpha) reaction in the boron dissolved in coolant water, and (3) ^{6}Li (n,alpha) reaction with the lithium—containing 0.1 percent ^{6}Li—in coolant water.(14,17) The calculated generation rate by fission (see Appendix C.1) is 210 µCi/s or 6,600 Ci/yr. Its production from boron and lithium computed for a station at a power level of 1473 MWt (which is the equivalent at Haddam Neck to full power at 0.8 capacity factor) is approximately 400 and 30 Ci/yr, respectively.(18) Accordingly, of the tritium discharge of 5,800 Ci during 1971, the first full year of operation after the initial core had been partially replaced, 5,400 Ci would be from fission. This suggests that approximately 80 percent of the tritium formed by fission during the year had leaked through the stainless-steel cladding. The actual leakage fraction would be somewhat less because some additional tritium from previous years would have accumulated in fuel elements remaining from core I.

Concentrations of tritium in the coolant (see Appendix B.1) were generally higher after the first refueling in May-June 1970 than before refueling. This may have been due to the new fuel elements or operation at higher power levels. Calculations for the first core had indicated that only approximately two-thirds of the fission-produced tritium had leaked through the cladding.(18) Immediately after refueling, and tapering off during a 2-week period, a much higher level of tritium than usual was observed in the coolant; this is attributed to a higher leakage rate caused by redistribution of power and temperature in the fuel elements remaining from the preceding core.(18) If, according to the annual tritium production values given above, almost all of the tritium in coolant water is

Table 2.1

Radionuclide Concentration in Reactor Coolant Water, μCi/ml*

Radionuclide		July 24, 1970	November 20, 1970	March 16, 1971
		from fuel		
12.3 -yr	^{3}H†	6.0	3.5	4.5
50.5 -d	^{89}Sr	1.0x10^{-5}	5.0x10^{-6}	4.5x10^{-5}
28.5 -yr	^{90}Sr	3 x10^{-7}	1.6x10^{-6}	2.7x10^{-6}
65 -d	^{95}Zr†	1.1x10^{-6}	8 x10^{-7}	1.9x10^{-5}
35.1 -d	^{95}Nb†	1.5x10^{-6}	2 x10^{-7}	1.0x10^{-5}
66.2 -hr	^{99}Mo†	NA	2.0x10^{-5}	4.7x10^{-4}
39.6 -d	^{103}Ru	<1 x10^{-7}	1.3x10^{-6}	1.1x10^{-5}
8.06-d	^{131}I	8.9x10^{-3}	2.4x10^{-2}	3.7x10^{-2}
20.9 -hr	^{133}I	NA	3.3x10^{-2}	7.2x10^{-2}
6.7 -hr	^{135}I	NA	2.5x10^{-2}	5.6x10^{-2}
5.29-d	^{133}Xe	8.8x10^{-2}	1.1	5.5x10^{-1}
2.3 -d	^{133m} Xe	NA	9.3x10^{-3}	1.8x10^{-2}
9.1 -hr	^{135}Xe	NA	2.8x10^{-2}	7.2x10^{-2}
2.07-yr	^{134}Cs†	6.2x10^{-5}	2.7x10^{-4}	7.7x10^{-4}
13 -d	^{136}Cs†	7.6x10^{-6}	8.2x10^{-5}	2.4x10^{-4}
30 -yr	^{137}Cs	8.5x10^{-5}	2.9x10^{-4}	7.2x10^{-4}
12.8 -d	^{140}Ba	1.0x10^{-5}	3 x10^{-6}	1.0x10^{-5}
32.4 -d	^{141}Ce	<1 x10^{-7}	~1.0x10^{-6}	2.3x10^{-5}
284 -d	^{144}Ce	<1 x10^{-7}	<2 x10^{-6}	2.2x10^{-5}
2.34-d	^{239}Np	NA	2.7x10^{-5}	5.2x10^{-4}
	gross alpha	<3 x10^{-9}	2 x10^{-8}	4.9x10^{-7}
		from activation of water, cladding, and construction materials		
5730 -yr	^{14}C	9.7x10^{-6}	3.4x10^{-5}	NA
15.0 -hr	^{24}Na	NA	9.0x10^{-4}	1.0x10^{-3}
14.3 -d	^{32}P	4.2x10^{-6}	1.1x10^{-5}	NA
88 -d	^{35}S	<1 x10^{-7}	1.2x10^{-6}	NA
27.7 -d	^{51}Cr	3.5x10^{-5}	1.0x10^{-5}	2.2x10^{-5}
313 -d	^{54}Mn	1.3x10^{-3}	1.3x10^{-5}	2.0x10^{-6}
2.7 -yr	^{55}Fe	3.9x10^{-3}	1.3x10^{-4}	<2 x10^{-7}
44.6 -d	^{59}Fe	4.0x10^{-5}	4.2x10^{-6}	<2 x10^{-7}
270 -d	^{57}Co	1.6x10^{-5}	5 x10^{-7}	<4 x10^{-7}
71.3 -d	^{58}Co	3.6x10^{-3}	1.5x10^{-4}	7.1x10^{-5}
5.26-yr	^{60}Co	5.6x10^{-4}	1.8x10^{-5}	5.5x10^{-6}
92 -yr	^{63}Ni	4.9x10^{-5}	NA	NA
60.2 -d	^{124}Sb†	1.6x10^{-6}	2 x10^{-7}	<1 x10^{-7}
115 -d	^{182}Ta	<1 x10^{-7}	1.5x10^{-6}	<3 x10^{-6}

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* Concentration at time of sampling; water at standard temperature and pressure.

† ^{3}H is also an activation product; ^{95}Zr, its daughter ^{95}Nb, and ^{99}Mo may also be activation products; ^{134}Cs and ^{136}Cs are produced by (n,gamma) reactions with fission-produced ^{133}Cs and ^{135}Cs, respectively, and ^{124}Sb may also be produced by (n,gamma) reactions with fission-produced ^{123}Sb.

Notes:

1. NA = not analyzed
2. < values are 3 sigma counting error.
3. The following fission products were not detected (usually <1x10^{-6} μCi/1): ^{91}Y, ^{97}Zr, ^{106}Ru, ^{127}Sb, ^{129}I, ^{132}Te, ^{143}Ce, ^{147}Nd. The radionuclides ^{65}Zn, ^{110m}Ag, ^{183}Ta, and ^{187}W were also not observed at this minimum detectable level.

from the fuel, then the major variations in the tritium concentration would arise from refueling, sudden power changes, and changes in the turnover rate of coolant water.

2.3.3 *Fission products in coolant water.* The measured concentrations of ^{131}I on the three occasions recorded in Table 2.1 were reasonably consistent with average monthly values from Haddam Neck operating reports (see Appendix B.1); $^{131}I/^{133}I$ atom ratios were also in agreement:

	July 1970	Nov. 1970	March 1971
	^{131}I concentration, µCi/ml		
this report	8.9×10^{-3}	2.4×10^{-2}	3.7×10^{-2}
operator's report	4.6×10^{-3}	2.8×10^{-2}	3.0×10^{-2}
	$^{131}I/^{133}I$ atomic ratio		
this report	---	6.7	4.8
operator's report	4.6	5.3	3.7

The concentrations of the radioiodine and many of the other fission products listed in Table 2.1 are consistent with calculated radionuclide release rates from the fuel when the fraction of fuel elements that leak radioactivity is 2×10^{-4}, as reported by the operator.(3) The fuel release rate, R_f (in µCi/s), is the product of the accumulation in fuel, A (in µCi, see Appendix C.1, last column), the escape rate coefficient, E, for the element(4) (in s^{-1}, see footnote to Appendix C.2), and the above mentioned fraction, F, of fuel elements releasing radioactivity:

$$R_f = AEF \qquad (2.1)$$

The calculated values of R_f are given in column 3 of Appendix C.2. At equilibrium, they are equal to the rates at which the radionuclides leave coolant water. Thus, the concentration of a radionuclide, C_r (in µCi/g), in the reactor coolant water is related to R_f by:

$$C_r = R_f/(v_r)(\Sigma\lambda_r) \qquad (2.2)$$

where

v_r : amount of reactor coolant water (1.6×10^8 g)

$\Sigma\lambda_r$: sum of radionuclide turnover coefficients in reactor coolant water (s^{-1}).

The turnover coefficients summed in column 4 of Appendix C.2, are as follows:

(1) The average coolant water loss reported by the operator was 2,200 kg/day, hence the water turnover coefficient, $\lambda_{water\ turnover}$ = (2,200 kg/ day)/(1.6×10^5 kg x 8.64×10^4 s/day) = $1.6 \times 10^{-7}\ s^{-1}$.

(2) Radioactive decay is characterized by the decay constants, λ_d, from column 3 of Appendix C.1 that range from 3×10^{-5} to $8 \times 10^{-10}\ s^{-1}$.

(3) The removal of ions by the demineralizer is the product of the removal fraction, f_d, and the water flow through the demineralizer relative to the total amount of coolant water, hence the turnover coefficient, $\lambda_{demineralizer}$ = (300 kg/min x f_d)/ (1.6×10^5 kg x 60 s/min) = $2.8 \times 10^{-5}\ s^{-1}$ when f_d is 0.9 (for all ions except cesium)(4) or $1.6 \times 10^{-5}\ s^{-1}$ when f_d is 0.5 (for cesium).

Radionuclides may also be removed from water by processes such as surface deposition or volatilization. The concentration values calculated in Appendix C.2 will be too high whenever such additional removal processes occur.

For a few radionuclides, these computed concentrations in Appendix C.2 are considerably different than the measured values in Table 2.1. Computed values are higher for ^{95}Nb, radioxenon, ^{134}Cs, and ^{137}Cs, and lower for ^{124}Sb and ^{144}Ce. Most of the measured ^{124}Sb probably was formed by neutron activation (see Table 2.1). Radioxenon isotopes are volatile (see Section 3.3.1). In view of the approximate nature of the values of A, E, F, $\lambda_{water\ turnover}$, and f_d on which the calculations are based, the agreement with many of the measured values suggests that the calculations yield useful estimates of reactor coolant concentrations.

The ratios of the measured concentrations to the computed values in Appendix C.2 were similar for the three iodine isotopes of widely different half lives:

	Measured concentrations/computed concentrations	
Radioiodine	Nov. 1970	March 1971
^{131}I	1.1	1.7
^{133}I	1.0	2.2
^{135}I	1.1	2.5

This suggests that the iodine isotopes in the coolant are in an "equilibrium mixture" that is formed when the passage of the isotopes from the fuel to the coolant is slow compared to the longest half-life of the isotopes (8 days, in this case).

Radioisotopes of tellurium (e.g., ^{132}Te) were not detected in the reactor water samples, possibly because tellurium is retained within the cladding or immediately removed from coolant water by deposition on surfaces. For ^{99}Mo, which was observed to be anionic (see Section 2.2.2), the ion-exchange removal fraction of 0.9 that was applied for other anions yielded computed values consistent with measured concentrations, although surface deposition ("plateout") had been indicated to be the major removal mechanism.(4)

2.3.4 *Activation products in coolant water.* The measured concentrations of the activation products in Table 2.1 were generally lower than had been calculated:

Radionuclide	Concentration in Reactor Coolant Water, µCi/liter		
	Model calculation(8)	Pre-operational calculation(1)	Measured
^{51}Cr	1.9	—	0.01–0.04
^{54}Mn	0.31	4.4	0.002–1.3
^{55}Fe	1.6	—	<0.0002–3.9
^{59}Fe	1	1.6	<0.0002–0.04
^{58}Co	6	8.5	0.07–3.6
^{60}Co	2	1.9	0.006–0.6

The highest measured concentrations of ^{54}Mn, ^{58}Co, and ^{60}Co approached the calculated values, however, and the measured concentration of ^{55}Fe exceeded the calculated value.

Following the observation of ^{14}C at Yankee-Rowe,(14) its annual production at PWR stations was estimated to be 30.4 Ci per 1,000 MWe in one report(19) and 6.3 Ci per 1,000 MWt in another.(20) According to these calculations, it is produced mostly by the reactions $^{14}N(n,p)$ and $^{17}O(n,alpha)$ in both the fuel and the coolant, while little ^{14}C is formed by $^{13}C(n,gamma)$ and ternary fission. The amount of ^{14}C generated at the Haddam Neck station at a power level of 1825 MWt and an 80-percent use factor would be either 14 or 9 Ci/yr.

2.3.5 *Radioactivity in secondary coolant water.* Many of the radionuclides observed in reactor coolant water were measured at concentrations lower by two orders of magnitude in blowdown water from the steam generators. These values, given in Table 2.2, were taken to represent concentrations in the 1.2 x 10^5 kg of steam generator water. The remaining 1.4 x 10^5 kg of secondary system water are steam and condensate. Radionuclide concentrations (except for ^3H) are believed to be lower in these then in reactor coolant water by factors of 1,000 (nonvolatile radionuclides) or 100 (volatile radionuclides in steam).(4)

The rate at which water leaks from the reactor coolant into the secondary coolant system at the steam generators can be determined by measuring radionuclide concentrations in the two systems and one other parameter. In analogy to equation 2.2, at equilibrium in the secondary system,

$$C_s = R_r/v_s \Sigma\lambda \qquad (2.3)$$

where C_s is the radionuclide concentration in steam generator water, R_r is the rate of radionuclide inleakage, v_s is the volume of water, and $\Sigma\lambda$ is the sum of the turnover coefficients of the radionuclide in the water. The rate of radionuclide inleakage can be taken to be the product of the radionuclide concentration in the reactor coolant, C_r, and the rate of leakage of water from the reactor coolant to the secondary coolant system, w_r. The value of $\Sigma\lambda$ for nonvolatile radionuclides is the sum of the radioactive decay constant, λ_d, and w_s/v_s, where w_s is the water discharge rate. Thus,

$$\frac{C_s}{C_r} = \frac{w_r}{w_s + (v_s)(\lambda_d)} \qquad (2.4)$$

As defined in Appendix C.3, v_s and w_s refer respectively to the water volume and blowdown rate at the steam generators for nonvolatile radionuclides. The same equation can be used for ^3H by applying the respective values for water in the entire secondary system and blowdown plus leakage rates from the entire system. The calculation is simplified if the half life of the measured radionuclide is either very long or very short, because $(v_s)(\lambda_d)$ or w_s will then be negligible relative to the other.

From ^3H measurements in the two systems on three occasions and water discharge rates from the secondary system reported by station staff,(21) the following reactor-to-secondary leakage rates, w_r, were computed with equation 2.4:

Sampling date	C_s/C_r	w_s, kg/day	w_r, kg/day
July 23/24, 1970	$2.3x10^{-3}$	55,000	130
Nov. 20, 1970	$3.4x10^{-3}$	53,000	180
Mar. 15/16, 1971	$4.9x10^{-3}$	64,000	310

The ^3H concentrations are from Tables 2.1 and 2.2 on the indicated dates; values of w_s are the average 38,000-kg/day leakage rate from the secondary system (Section 2.1.5) plus the daily blowdown rate (Appendix C.3). The first inleakage rate lies within the range of 75–150 kg/day reported by the station operator,(4) while the other two are somewhat higher. Possible sources of error are the use of an average leakage rate for the secondary system because the specific values for these days were not available, and sample collection on two occasions with a one-day interval instead of simultaneously.

The concentrations of the nongaseous radionuclides in steam generator water, calculated by equation 2.4 from the reactor coolant concentrations in Table 2.1 and the above-listed reactor-to-secondary leakage rates, are given in Appendix C.3. The calculated values for November 20, 1970 are all within a factor of two of the measured concentrations in Table 2.2. On March 15/16, three of the values—for ^{58}Co, ^{95}Zr, and ^{133}I—are different, and on July 23/24, almost every calculated value differs from the measured one. On these two occasions, also, some radionuclides predicted to be at measurable levels were not detected. Such differences can arise from fluctuations in the radionuclide concentrations and rates of water turnover, or from a sample that does not represent all four steam generators.

Table 2.2

Radionuclide Concentration in Steam Generator Water, µCi/ml

Radionuclide	July 23, 1970	Sept. 15, 1970	Nov. 20, 1970	Mar. 15, 1971	Mar. 16, 1971	April 14, 1971
^{3}H	1.4×10^{-2}	3.3×10^{-3}	1.2×10^{-2}	2.2×10^{-2}	NA	2.1×10^{-2}
^{14}C	$< 1 \times 10^{-7}$	$< 1 \times 10^{-7}$	2×10^{-7}	3×10^{-7}	NA	$< 1 \times 10^{-7}$
^{24}Na	NA	NA	NA	NA	3.0×10^{-6}	NA
^{32}P	3×10^{-7}	2.2×10^{-6}	$< 1 \times 10^{-7}$	1.4×10^{-6}	NA	1.9×10^{-6}
^{51}Cr	$< 1 \times 10^{-6}$	1.6×10^{-6}	$< 1 \times 10^{-6}$	$< 1 \times 10^{-6}$	NA	$< 1 \times 10^{-6}$
^{54}Mn	1×10^{-7}	1.1×10^{-6}	3×10^{-7}	$< 1 \times 10^{-7}$	$< 1 \times 10^{-7}$	1×10^{-7}
^{59}Fe	$< 1 \times 10^{-7}$	1.8×10^{-6}	1.9×10^{-6}	$< 1 \times 10^{-7}$	NA	$< 1 \times 10^{-7}$
^{58}Co	2×10^{-7}	2.0×10^{-5}	1.7×10^{-6}	$< 1 \times 10^{-7}$	1×10^{-7}	$< 1 \times 10^{-7}$
^{60}Co	1×10^{-7}	7×10^{-7}	2×10^{-7}	1×10^{-7}	1×10^{-7}	$< 1 \times 10^{-7}$
^{90}Sr	1×10^{-8}	$< 1 \times 10^{-8}$	1×10^{-8}	5×10^{-8}	NA	1×10^{-8}
^{95}Zr	1×10^{-7}	1×10^{-7}	$< 1 \times 10^{-7}$	$< 1 \times 10^{-7}$	NA	$< 1 \times 10^{-7}$
^{99}Mo	NA	9.1×10^{-6}	$< 2 \times 10^{-7}$	4.0×10^{-6}	1.8×10^{-6}	7.7×10^{-6}
^{131}I	8.0×10^{-5}	4.6×10^{-4}	1.2×10^{-4}	5.5×10^{-4}	3.3×10^{-4}	3.6×10^{-4}
^{133}I	NA	NA	8.9×10^{-5}	3.8×10^{-4}	3.3×10^{-4}	5.4×10^{-4}
^{135}I	NA	NA	NA	NA	1.4×10^{-4}	NA
^{134}Cs	2.6×10^{-6}	4.3×10^{-5}	3.0×10^{-6}	1.1×10^{-5}	8.1×10^{-6}	8.9×10^{-6}
^{136}Cs	5×10^{-7}	1.3×10^{-5}	6×10^{-7}	2.5×10^{-6}	1.6×10^{-6}	1.8×10^{-6}
^{137}Cs	3.0×10^{-6}	4.9×10^{-5}	4.0×10^{-6}	1.1×10^{-5}	8.0×10^{-6}	8.4×10^{-6}

Notes:

1. NA: not analyzed
2. < values are 3-sigma counting error
3. The following radionuclides were not detected: ($< 1 \times 10^{-7}$ µCi/ml) ^{55}Fe, ^{57}Co, ^{63}Ni, ^{89}Sr, ^{95}Nb, ^{110m}Ag, ^{140}Ba, ^{141}Ce, ^{182}Ta, and ^{185}W; ($< 5 \times 10^{-7}$ µCi/ml) ^{144}Ce, ^{239}Np; ($< 1 \times 10^{-9}$µCi/ml) gross alpha.

2.3.6 *Radionuclides in refueling cavity water.* While the reactor was shut down for refueling (April 16 to May 21), a sample of refueling cavity water was analyzed radiochemically with the results shown in Table 2.3. All radionuclides except ^{3}H and relatively short-lived ones such as ^{99}Mo and ^{131}I were at higher concentrations in this water than in the most recent reactor coolant sample (Table 2.1, last column). Radioisotopes of cobalt and iron were at particularly high concentrations relative to reactor coolant water, and ^{106}Ru and ^{110m}Ag were found in this sample although below the detection limit in reactor coolant water.

2.4 References

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Table 2.3

Radionuclide Concentration in Refueling Cavity Water on May 7, 1971

Radionuclide	Concentration, μCi/ml	Radionuclide	Concentration, μCi/ml
^{3}H	3.6×10^{-2}	^{95}Nb	1.4×10^{-4}
^{14}C	3×10^{-7}	^{99}Mo	1.3×10^{-5}
^{32}P	1.2×10^{-4}	^{103}Ru	6.8×10^{-5}
^{51}Cr	1.6×10^{-3}	^{106}Ru	2.0×10^{-5}
^{54}Mn	1.5×10^{-4}	^{110m}Ag	1.4×10^{-5}
^{55}Fe	1.4×10^{-3}	^{124}Sb	6.8×10^{-5}
^{59}Fe	1.8×10^{-4}	^{131}I	2.3×10^{-3}
^{57}Co	3.6×10^{-5}	^{134}Cs	6.4×10^{-3}
^{58}Co	1.4×10^{-2}	^{136}Cs	5.2×10^{-4}
^{60}Co	8.5×10^{-4}	^{137}Cs	5.5×10^{-3}
^{89}Sr	1.0×10^{-4}	^{140}Ba	7.8×10^{-5}
^{90}Sr	1.6×10^{-5}	^{141}Ce	1.5×10^{-4}
^{95}Zr	8.9×10^{-5}	^{144}Ce	2.0×10^{-4}

Notes:

1. Concentrations at time of sampling.
2. Reactor was shut down for refueling on April 16, 1971.

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3. AIRBORNE RADIOACTIVE DISCHARGES

3.1 Gaseous Waste System and Samples

3.1.1 *Gaseous waste system.* The airborne radionuclides from the reactor system are removed principally by radioactive decay and by discharge to the atmosphere. The airborne radionuclides are either gaseous—fission-produced tritium, krypton, xenon, and iodine and activation-produced tritium, carbon, and argon—or on particles. The pathways for gaseous waste are depicted in Figure 3.1.(1-10) A program to augment the system for reducing the amounts of radioactive waste effluents is now under way.(8)

Gases separate from reactor coolant water after discharge through the chemical and volume control system (CVCS) and when the reactor-vessel cover is removed after shutdown. Haddam Neck reports that the reactor is opened once each year for refueling,(8) during which the gases are vented to the vapor container. Gases from the CVCS are transferred directly through the head space in the primary drain tank to the main waste gas system. Flow proceeds in cascade through the head space in the first and second boron waste storage tanks to a 570-m^3 waste gas surge sphere. A polyurethane-coated nylon cloth diaphragm ("Wiggins balloon") attached at the sphere equator restores waste gas to the boron waste storage tanks as liquid drains from them.

When the surge sphere is full, a portion of the stored gas is pumped at a constant rate between 0.03 and 0.3 m^3/min through a glass fiber filter (for removing particles) to a plenum where the gas is diluted before reaching the primary vent stack. Analyses of stored gas by station staff indicate that its main radioactive constituent is ^{133}Xe, with some ^{85}Kr and ^{135}Xe. From January 1968 to December 1970, 28- to 210-m^3 volumes of gas were released on eight occasions. The average discharge was 100 m^3.(4) No routine releases are reported for 1971, but gas was released in March and April of 1971 in connection with this study to test methods for measuring this effluent in the environment (see Section 6). Leakage, if any, through the sphere diaphragm is exhausted continuously to the stack duct.

Leaks in valve stems, pump shaft seals and other system equipment allow some reactor coolant and associated gases to enter the atmospheres of the vapor container and the primary auxiliary building. The station has reported that these coolant leakage rates to the two buildings are negligible. The Environmental Statement(8) assumed leakage rates during routine operation of 110 kg/day (40 gal/day at a specific gravity of 0.7) to the vapor container and 75 kg/day (20 gal/day) to the auxiliary building.(8) These are also the rates estimated for "model" PWR plants by the AEC(10) and EPA.(11)

Airborne radioactivity accumulates and decays in the vapor container because its air is exhausted only for refueling or major maintenance. From January 1968 through December 1971, the 63,000-m^3 structure was purged four times for repairs and twice for refueling.(4) The air is exhausted to the discharge plenum initially at 1,000 to 2,000 m^3/min. When radioactivity levels become lower, the exhaust rate is reduced; it was, for example, 140 m^3/min to remove gases from the open reactor vessel during the April-May 1971 refueling operation.(3)

Gases that leak from reactor coolant into the primary auxiliary building are removed continuously by the building ventilation system. Air is exhausted at the rate of 570 m^3/min (20,000 cfm) to the discharge plenum. In addition, approximately 30 cc of gas from sampling and analyzing reactor coolant are discharged weekly through a hood to the plenum.(3) These direct discharges of reactor-system gases contain short-lived radionuclides.

Gases from two aerated drain tanks that collect leaking reactor coolant water from the vapor container and from hoods are discharged through headers as the tanks fill, and pass through a filter to the plenum. The gas volume is negligible, an estimated 6 m^3/week.(3) Small amounts of off-gas from the boron recovery evaporators are also processed to the plenum.

When reactor coolant water leaks into the secondary coolant system at the steam generators (see Section 2.3.5), it is accompanied by radioactive gases. Noncondensable gases are removed from the secondary

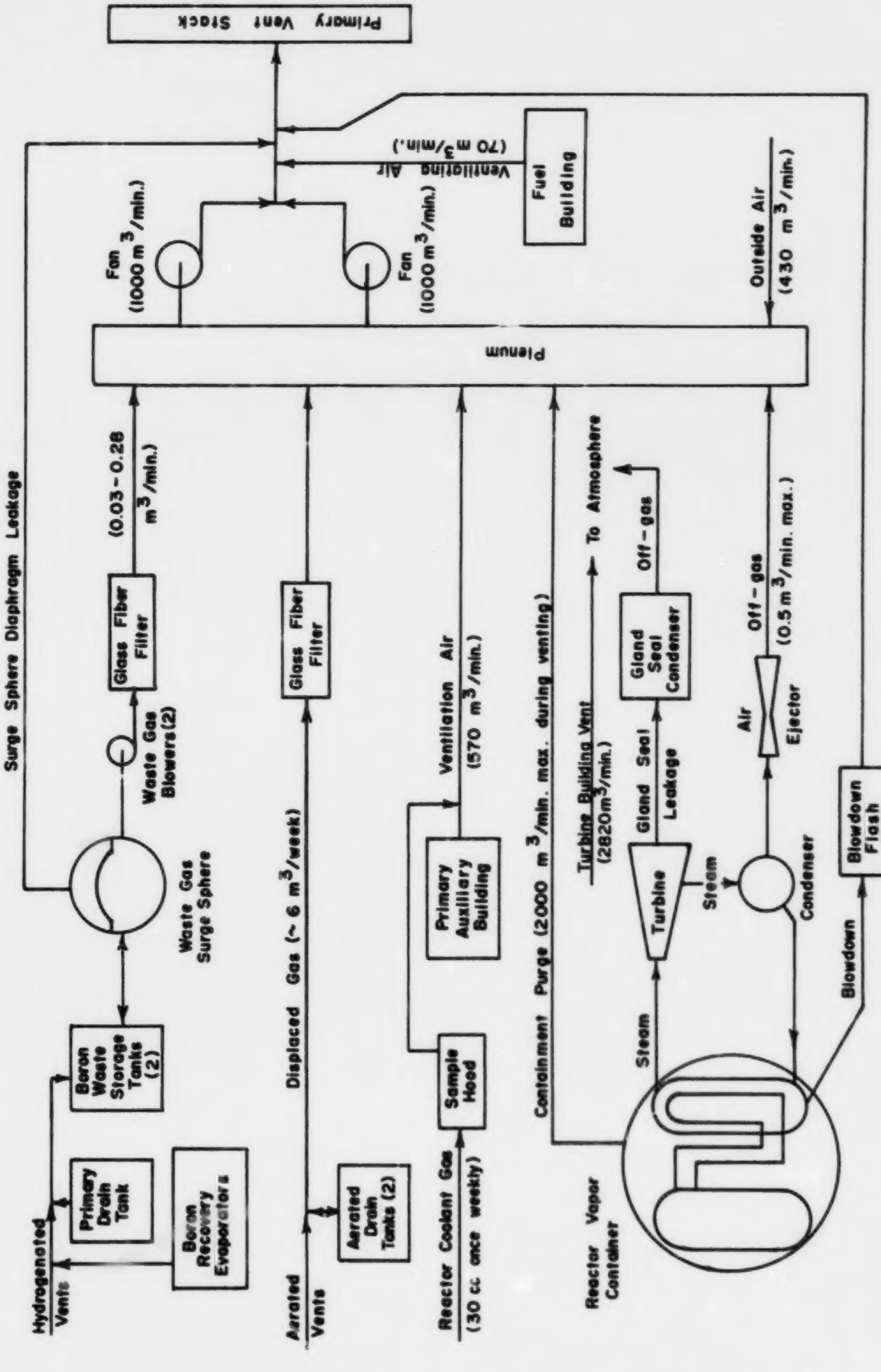
Primary Vent Stack
Surge Sphere Diaphragm Leakage
(0.03 - 0.28 m³/min.)
Fan (1000 m³/min.)
Fan (1000 m³/min.)
Ventilating Air (70 m³/min.)
Fuel Building
Plenum
Outside Air (430 m³/min.)
Hydrogenated Vents
Primary Drain Tank
Boron Recovery Evaporators
Boron Waste Storage Tanks (2)
Waste Gas Surge Sphere
Waste Gas Blowers (2)
Glass Fiber Filter
Aerated Vents
Aerated Drain Tanks (2)
Displaced Gas (~ 6 m³/week)
Glass Fiber Filter
Reactor Coolant Gas (30 cc once weekly)
Sample Hood
Primary Auxiliary Building
Ventilation Air (570 m³/min.)
Containment Purge (2000 m³/min. max. during venting)
Reactor Vapor Container
Steam
Turbine
Turbine Building Vent (2820 m³/min.)
To Atmosphere
Gland Seal Leakage
Gland Seal Condenser
Off-gas
Steam
Condenser
Condenser
Air
Ejector
Off-gas (0.5 m³/min. max.)
Blowdown
Blowdown Flash Tank

Figure 3.1 Gaseous Waste Disposal System

coolant at the main condenser by a steam jet air ejector (SJAE) at a rate between 0.2 and 0.5 m^3/min. Off-gas is discharged to the plenum and monitored continuously by a Geiger-Mueller (G-M) radiation detector. This is expected to be the major pathway for releasing short-lived radioactive gases to the environment.

Gaseous waste discharged to the plenum is diluted by outside air and by ventilation air exhausted from the primary auxiliary building. A fan operating at 1,000 m^3/min blows the air through a discharge duct to the 1.8-m-dia., 53-m-high cylindrical primary vent stack. A second fan is operated for releasing gases with higher radioactivity concentrations, such as surge sphere contents. Radiation levels in the duct are monitored continuously by a flow-through G-M detector. A stream of this gas is withdrawn isokinetically at a rate of approximately 50 liters/min and passed through two sample collectors—a glass fiber filter followed by a charcoal-impregnated filter. These are changed daily and analyzed for radioactive particles and radioiodine.(9)

Three other continuous pathways for gaseous radionuclides from the secondary coolant to the environment are known:

1) Air inleakage to the turbine is prevented by passing 0.1 percent of the steam through the shaft gland seal annulus. The steam is then condensed and returned to the system. Noncondensable gases are removed by an air ejector at the condenser and vented to the atmosphere through a small stack atop the turbine building, 23 m above ground.

2) Leaks from many small sources occur in the turbine building. Haddam Neck has reported that leakage to the building is negligible.(8) For calculating radionuclide discharges, the steam leakage rate was assumed to be 9,300 kg/d, one-half the rate (1,700 lb/hr) for the "model" 3,500-MWt PWR system.(10) The building air is discharged through a roof vent at the rate of 2,800 m^3/min (100,000 cfm),(9) and also reaches the outside through doorways and other openings.

3) Steam generator blowdown water is pumped to a flash tank where approximately 35 percent flashes as steam that is discharged directly to the primary vent stack at an average rate of 6,700 kg/day (see Section 2.1.5). The flash tank discharge is considered to be the major pathway of radioiodine to the atmosphere.(8)

A minor source in the secondary system is the pumping of gases from the condenser to establish a vacuum for start-up.(10) Steam discharged to the atmosphere at relief valves during abnormal operations has also been mentioned as an occasional effluent.(11)

Radioactive gases from fuel pool water in which used fuel elements are stored in the fuel building diffuse into the building atmosphere. Ventilation air from the fuel building is exhausted continuously to the primary vent stack at a rate of 70 m^3/min (2,500 cfm).

3.1.2 *Radionuclide release.* Radioactive gases discharged by Haddam Neck are limited by the AEC as follows:(12)

> When averaged over any calendar year, the release rate of radioactivity consisting of noble gases and other isotopes with half lives less than eight days discharged at the plant stack shall not exceed 3 x 10^4 x (MPC) curies per second, where MPC is the value in microcuries per cubic centimeter given in Appendix B, Table II, Column 1 of 10 CFR 20. The maximum release rate when averaged over any one hour shall not exceed 10 times the yearly averaged limit.
>
> At any time when the averaged release rate for a week exceeds 30 percent of the annual average limit given above, the licensee shall make provisions for sampling iodine-131 to assure that its release rate averaged over any calendar year does not exceed 66 x (MPC) curies per second.

The values in the cited Table II are derived from Section 20.105 of 10 CFR 20,(13) which limits the added radiation dose to an individual in an unrestricted area to 500 mrem/yr. The permissible limits given in Table II, Column 1 of 10 CFR 20 have been increased by a factor of 1000 by the AEC in consideration of atmospheric dilution of effluents before reaching the site exclusion boundary.(1) The limits of average annual effluent concentrations and annual releases allowed to the Haddam Neck station for individual radionuclides are as follows:

Radionuclide		Effluent concentration limit, uCi/cc	Annual release limit,* Ci
Gases			
12.3 -yr	^{3}H (as HT)	4 x 10^3	2.1 x 10^7
	(as HTO)	2 x 10^4	1.1 x 10^5
5730 -yr	^{14}C (s)	1 x 10^4	5.3 x 10^4
	(as CO$_2$)	1 x 10^3	5.3 x 10^5
1.83-h	^{41}Ar	4 x 10^5	2.1 x 10^4
4.48-h	^{85m}Kr	1 x 10^4	5.3 x 10^4
10.7 -yr	^{85}Kr	3 x 10^4	1.6 x 10^5
1.27-h	^{87}Kr	2 x 10^5	1.1 x 10^4
2.80-h	^{88}Kr	2 x 10^5	1.1 x 10^4
11.9 -d	^{131m}Xe	4 x 10^4	2.1 x 10^5
2.25-d	^{133m}Xe	3 x 10^4	1.6 x 10^5
5.29-d	^{133}Xe	3 x 10^4	1.6 x 10^5
9.15-h	^{135}Xe	1 x 10^4	5.3 x 10^4
Other fission gases, half-lives <2 hr		3 x 10^5	1.6 x 10^4

Particles and Radioiodines			
313 -d	^{54}Mn (s & i)	1 x 10^{-9}	5.3 x 10^{2}
2.7 -yr	^{55}Fe (s & i)	3 x 10^{-8}	1.6 x 10^{4}
71.3 -d	^{58}Co (i)	2 x 10^{-9}	1.1 x 10^{3}
5.26-yr	^{60}Co (i)	3 x 10^{-10}	1.6 x 10^{2}
17.8 -m	^{88}Rb (i)	3 x 10^{-8}	1.6 x 10^{4}
50.5 -d	^{89}Sr (s)	3 x 10^{-10}	1.6 x 10^{2}
28.5 -yr	^{90}Sr (s)	3 x 10^{-11}	1.6 x 10^{1}
8.06-d	^{131}I (s)	1 x 10^{-10}	5.3 x 10^{1}
20.9 -h	^{133}I (s)	4 x 10^{-10}	2.1 x 10^{2}
2.07-yr	^{134}Cs (i)	4 x 10^{-10}	2.1 x 10^{2}
30.0 -yr	^{137}Cs (i)	5 x 10^{-10}	2.6 x 10^{2}

*Based on a continuous stack discharge of 1.66 x 10^7 cc/s (5.25 x 10^{14} cc/yr).

Notes:

1. The individual limits apply in the absence of other radionuclides; if several radionuclides are present, the sum of individual percentages of the limit may not exceed 100.
2. s = soluble; i = insoluble.

The station has reported the following annual airborne releases between 1967 and 1973:(6,7)

Year	Noble and activation gases, Ci	Halogens and particles, Ci	^{3}H, Ci	Fraction of allowable, %*
1967	0.02	0.001	--	--
1968	3.7	Negligible	9.0	0.006
1969	190	0.0004	2.52	0.07
1970	641	0.001	0.082	0.25
1971	3251	0.231	0.88	1.35
1972	645	0.018	6.55	1.47
1973	31.8	0.029	50.61	0.01

*Based on allowable concentrations of individual radionuclides measured in plant effluent.

Monthly values reported by the station of volume and radioactivity of gaseous effluent are tabulated in Appendices B.2, B.3, and B.4. Volumes up to 10^4 m^3 per month reflect routine discharges and occasional stored gas releases. Greater volumes indicate (except in September 1971) that the vapor container was purged. Higher releases of radioactivity are usually associated with discharges of stored gas. Individual radionuclides are given in Appendices B.3 and B.4, and their sources are identified in B.4. Data are included for 17.8-min ^{88}Rb, 20.9-h ^{133}I, and 15.6-min ^{135m}Xe, which, because of their relatively short half-lives or small quantities were not measured in this study.

3.1.3 *Sample collection.* Samples of gas flashed during reactor coolant sampling were collected in duplicate on November 20, 1970, and February 9, 1971. Aliquots were contained at atmospheric pressure in 9-cc serum bottles sealed with rubber stoppers held by crimped aluminum holders. Off-gas discharged through the exhaust line from the SJAE was collected in 1.8-liter metal bottles on eight occasions from July 1970 to April 1971.

The contents of the gas surge sphere were sampled on July 27, 1970, and April 14, 1971, and on three occasions just before stored gas was discharged. These releases were conducted under the following conditions:

Date (sampling time)	Release interval, hrs	Discharge rate, m^3/min	Volume, m^3
Sept. 16, 1970			
(0845 hrs)	1152–1530	0.100	21.8
March 15, 1971			
(1005 hrs)	1100–1340	0.142	48.7
	1340–1530	0.227	
April 16, 1971			
(0745 hrs)	0845–1100	0.113	27.2
	1100–1200	0.198	

Samples were collected at atmospheric pressure in 100- or 850-cc bottles. Aliquots were transferred to 12-cc glass bottles with rubber stoppers or 9-cc glass bottles sealed with rubber stoppers held by crimped aluminum holders.

Primary vent stack effluents were sampled with an air pump connected to a single-nozzle probe, centered in the stack discharge duct, that is routinely used for stack monitoring. Gaseous emissions were collected in 8.2-liter evacuated metal bottles on September 15, 1970, March 16, 1971, and April 14, 1971, during routine discharges and at the times of stored gas releases. Particulate emissions were sampled by means of a Unico filter holder for 5-cm-dia MSA 1106 glass fiber or HV-70 particulate filters. Behind the filter, a 3.2-cm-dia container for 26 g bed of activated charcoal from Cesco type B cartridges was inserted for sampling gaseous radioiodine. A second 26-g bed of activated charcoal and a 62-g bed of KI-impregnated charcoal (Mine Safety Appliances type 85851) were placed behind the first bed on one occasion to observe the collection efficiency of the sampler, but no additional information was obtained because all of the charcoal was inadvertently combined for analysis. Flow rates varied from 40 to 50 liters/min. Samples were obtained July 27 to August 4, 1970, September 15–16, 1970, March 15–16, 1971, and April 14–15, 1971. Separate samples were obtained while gas from the waste surge sphere was released. In addition, 3 gaseous radioiodine samples were obtained on June 1–4, 1971, shortly after start-up with core III.

Vapor container atmosphere was sampled by filling evacuated 8.2-liter containers inside the structure near the personnel access lock. On November 20, 1970, the ambient container temperature was 29° C and the relative humidity was 40 percent of saturation; on March 16, 1971, they were 35° C and 21 percent. One liter of atmospheric moisture condensate was obtained on November 20, 1970, and 100 cc, on March 16, 1971. The latter sample was collected from the condensate drain. Another air sample was obtained on May 5, 1971, when the vapor container building was open to the outside during refueling.

Gas samples of 8.2-liter volumes were collected in three other buildings to measure ambient radionuclide concentrations: 1) turbine hall, on April 15, 1971, near the ventilation air exhaust grate; 2) fuel storage building, on February 9, 1971, on the upper level near the spent fuel pit; and 3) primary auxiliary building, on February 9, 1971, in the corridor outside the sampling hood area. Samples could not be collected in ventilation discharge ducts.

Various difficulties prevented sampling of flashed steam generator blowdown, off-gas from the condenser air ejector for the turbine gland seals, gas from the boron recovery evaporators, and air vented from the aerated liquid waste tanks.

3.2 Analysis

3.2.1 *Gamma-ray spectrometry.* Analytical measurement systems and procedures were similar to those described in Section 3.2.1 of the study at the Yankee-Rowe station.(14) Xenon-133 values are based on results of detector efficiency calibrations conducted with the ^{133}Xe standard issued by the National Bureau of Standards in October 1973. Photon intensities for all krypton and xenon radionuclides were taken from a recent data summary.(15) Samples obtained on February 9, 1971, were analyzed by NaI(Tl) and Ge(Li) detector systems (see Fig. 3.2) within 4 to 6 hours after collection to measure short-lived radionuclides.*

Radioiodine adsorbed on charcoal was analyzed with the NaI(Tl) detector system. The results were corrected for 92-percent collection efficiency.(16) This has been confirmed by observations in measuring stack gases at BWR nuclear power stations, where efficiencies of 88 and 90 percent were observed(17,18) although most of the ^{131}I was in an organic form(18) such as methyl iodide.

3.2.2 *Radiochemical analysis.* Most samples were analyzed for ^{3}H, ^{14}C, ^{85}Kr, and radiostrontium as described in the Yankee-Rowe report,(14) with the modifications indicated below. Samples containing low concentrations of ^{85}Kr were analyzed only by using plastic scintillator spheres. Beginning with the gas samples obtained on February 9, 1971, the tritiated moisture fraction was separated at −76° C in a freeze trap located at the beginning of the ^{3}H-^{14}C gas analysis train. A bubbler containing $Ba(OH)_2$ was inserted behind the trap to collect the $^{14}CO_2$ fraction. The remaining gases were then catalytically oxidized and the resulting H_2O and CO_2 were collected in another freeze trap and $Ba(OH)_2$ bubbler, respectively. The chemical forms of ^{14}C in the non-CO_2 fractions and of gaseous ^{3}H have not been identified.

3.3 Results and Discussion

3.3.1 *Radioactive gases in reactor coolant.* All krypton and xenon radioisotopes with half-lives longer than one hour that are produced at high yields by fission were measured in gas released from reactor-coolant samples, as shown in Table 3.1. Tritium and the activation products ^{14}C and ^{41}Ar were also detected. Tritium was found as a gas (not in water vapor) and ^{14}C was in a chemical form other than CO_2. Argon-41, formed by the neutron activation of argon in air, was barely detectable, as expected because the reactor coolant is de-aerated before operation.(3)

The station reported the following noble gas measurements:(6)

Date, 1970	Concentration in gas, µCi/cc	
	^{133}Xe	^{135}Xe
Sept. 4	50.7	3.05
Oct. 30	30.	1.61
Nov. 20	35.8	2.65

The values on Nov. 20 are comparable to those given for ^{133}Xe and ^{135}Xe in Table 3.1.

The concentrations of the measured radionuclides would be as follows in reactor coolant water at a gas concentration of 35 cc/kg water (see Section 2.1.2):

Radionuclide	Concentration in reactor coolant water based on gas-phase measurements, µCi/g water	
	Nov. 20, 1970	Feb. 9, 1971
^{85m}Kr	—	4.6×10^{-2}
^{85}Kr	1.3×10^{-2}	2.4×10^{-2}
^{87}Kr	—	6.0×10^{-2}
^{88}Kr	—	8.8×10^{-2}
^{133m}Xe	1.8×10^{-2}	2.4×10^{-2}
^{133}Xe	1.1	1.8
^{135}Xe	6.0×10^{-2}	3.0×10^{-1}

Compared to the values measured directly in the water (see the Nov. 20, 1970, sample in Table 2.1), the ^{133}Xe concentration is identical while the concentrations of ^{133m}Xe and ^{135}Xe in the gas samples are twice as high. Radioxenon values measured in the water may be low because the sample, not being intended for gas analysis, was not maintained air-tight.

The concentrations of the shorter-lived xenon and krypton radioisotopes are reasonably consistent with the values computed in Appendix C.2. This suggests

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*We thank Messrs. Christopher Nelson and Gerald Karches, formerly with the Northeastern Radiological Health Laboratory, U. S. Public Health Service, for facilitating these analyses.

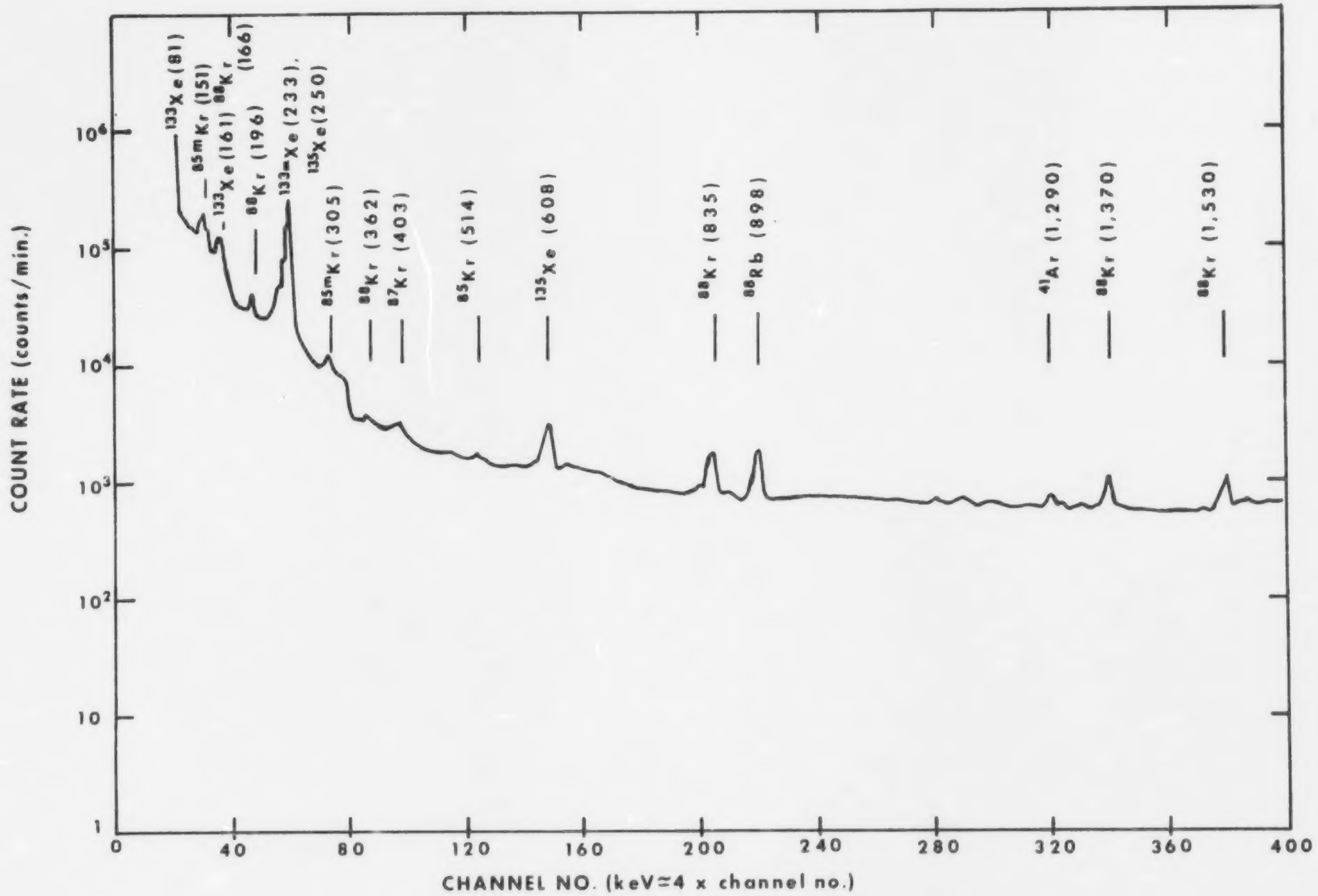

Figure 3.2 - Gamma-ray spectrum of off-gas from sampling reactor coolant, 0-1600 keV
Detector: Ge (Li) 10.4 cm^2× 11mm, trapezoidal.
Sample: 9cc bottle containing 1cc gas; collected Feb. 9, 1971 at 0755 hour.
Count: 10 minutes on Feb. 9, 1971 (1405-1415 hr) at Winchester, Mass.

Table 3.1

Radioactive Gases Released to Stack from Sampling Reactor Coolant

Radionuclide		Concentration, μCi/cc		Estimated annual release,* Ci
		Nov. 20, 1970	Feb. 9, 1971	
12.3 -yr	^{3}H (gas)	1.0 ± 0.1 x 10^{-5}	3.8 ± 0.9 x 10^{-5}	3.5 x 10^{-8}
5730 -yr	^{14}C (non-CO_2)†	1.7 ± 0.4 x 10^{-3}	3.2 ± 0.1 x 10^{-4}	1.4 x 10^{-6}
1.83-hr	^{41}Ar	NA	1.1 ± 0.8 x 10^{-1}	1.6 x 10^{-4}
4.48-hr	^{85m}Kr	NA	1.3 ± 0.1	1.9 x 10^{-3}
10.7 -yr	^{85}Kr	3.8 ± 0.1 x 10^{-1}	6.8 ± 0.2 x 10^{-1}	7.7 x 10^{-4}
76.3 -m	^{87}Kr	NA	1.7 ± 0.3	2.4 x 10^{-3}
2.80-hr	^{88}Kr	NA	2.5 ± 0.1	3.6 x 10^{-3}
2.25-d	^{133m}Xe	5.1 ± 0.1 x 10^{-1}	7.0 ± 0.4 x 10^{-1}	8.7 x 10^{-4}
5.29-d	^{133}Xe	3.2 ± 0.1 x 10^{1}	5.1 ± 0.1 x 10^{1}	6.0 x 10^{-2}
9.15-hr	^{135}Xe	1.7 ± 0.1	8.5 ± 0.6	7.3 x 10^{-3}

*Based on release of 30 cc of gas during weekly sampling operation for 48 weeks per year.

†The concentration of ^{14}C in CO_2 was < 1 x 10^{-6} μCi/cc on Nov. 20, 1970.

Notes:

1. ± values indicate analytical error expressed at 2-sigma; < values are minimum detectable levels at 3-sigma counting error.
2. NA – not analyzed.
3. ^{3}H as water vapor was not measured.

that the concentrations of the noble gas radionuclides that could not be measured because of their short half lives and/or low abundances—^{83m}Kr, ^{89}Kr, ^{131m}Xe, ^{135m}Xe, ^{137}Xe, and ^{138}Xe—are of the magnitude given in Appendix C.2.

The concentrations based on the measured gas values were, on the average, twice as high as the concentrations computed in Appendix C.2 for the short-lived krypton isotopes, one-half as high for the xenon isotopes, and one-eighth as high for ^{85}Kr. The differences for the short-lived noble gases may be due to the use of a mean escape rate coefficient in computing the leakage of both krypton and xenon from the fuel, when the escape rate is actually greater for the smaller krypton atoms. The relatively low measured concentration of ^{85}Kr may indicate that the gases leave the reactor coolant system more rapidly than inferred from the turnover of water according to the model discussed in Section 2.3.3. Of the measured noble gas radionuclides, only ^{85}Kr leaves the reactor coolant system mainly by discharge and leakage; the other noble gases are removed mostly by radioactive decay within the system.

The annual discharge of gaseous radionuclides during sampling estimated in Table 3.1 is a minute fraction of the total discharges summarized in Section 3.3.13. The unmeasured noble gas radionuclides listed above would add only approximately 0.001 Ci/yr to this total, as estimated from the annual discharge of 1440 cc of gas (see footnote to Table 3.1)—i.e., the gas in 41 kg reactor coolant water at the computed concentrations in Appendix C.2.

The concentration of ^{3}H in water vapor, although not measured in these samples, is expected to be at low but detectable levels. At a concentration of 3 μCi/g water (see Table 2.1 and Appendix B.1) when the two samples were collected, and a saturation water content of 2 x 10^{-5} g/cc in the gas samples at room temperature, the ^{3}H content would be 6 x 10^{-5} μCi/cc. The annual discharge, in 1,440 cc, is 9 x 10^{-8} Ci.

3.3.2 *Radionuclides in the waste gas surge sphere.* All of the longer-lived gaseous radionuclides in reactor coolant were also observed in the stored gas: ^{3}H, ^{14}C, ^{85}Kr, ^{133m}Xe, and ^{133}Xe. Their concentrations, shown in Table 3.2, were lower than in the reactor coolant by factors of approximately 5 for ^{85}Kr, 100 for ^{133}Xe, and 1000 for ^{133m}Xe. This reduction is expected from dilution by nitrogen purge and decay of the shorter-lived radionuclides.

Xenon-135 was not measurable at this laboratory (< 9 x 10^{-4} μCi/cc in the optimum sample) because of its low initial abundance and its decay by several half-lives before analysis. The Haddam Neck staff provided the following measurements:

Table 3.2

Radioactivity Contents of Waste Gas Surge Sphere

Radionuclide	Concentration, µCi/cc					Estimated annual release,* Ci
	July 27, 1970	Sept. 16, 1970	March 15, 1971	April 14, 1971	April 16, 1971	
^3H (gas)	$1.7 \pm 0.1 \times 10^{4}$**	$5 \pm 1 \times 10^{4}$**	NA	$1.6 \pm 0.4 \times 10^{4}$	$< 3 \times 10^{4}$	7.1×10^{3}
^3H (H_2O)	NA	NA	NA	$< 3 \times 10^{4}$	$< 2 \times 10^{4}$	$< 6 \times 10^{4}$
^{14}C (non-CO_2)	$7.3 \pm 0.2 \times 10^{5}$**	$1.4 \pm 0.1 \times 10^{4}$**	NA	$2.6 \pm 0.1 \times 10^{4}$	$1.1 \pm 0.1 \times 10^{4}$	3.2×10^{2}
^{14}C (CO_2)	NA	NA	NA	$< 2 \times 10^{4}$	$< 1 \times 10^{4}$	$< 3 \times 10^{4}$
^{85}Kr	$1.3 \pm 0.1 \times 10^{2}$	$1.1 \pm 0.1 \times 10^{1}$	$1.3 \pm 0.1 \times 10^{1}$	$1.2 \pm 0.1 \times 10^{1}$	$1.4 \pm 0.1 \times 10^{1}$	2.9×10^{1}
^{131m}Xe	$7 \pm 2 \times 10^{4}$	$2.1 \pm 0.2 \times 10^{1}$	$< 9 \times 10^{4}$	$7 \pm 5 \times 10^{4}$	$1.7 \pm 0.4 \times 10^{1}$	4×10^{1}
^{133}Xe	$1.8 \pm 0.1 \times 10^{1}$	1.1 ± 0.1	$7.2 \pm 0.1 \times 10^{2}$	$3.1 \pm 0.1 \times 10^{1}$	$4.4 \pm 0.1 \times 10^{1}$	1.3×10^{2}
^{135}Xe	NA	$< 9 \times 10^{4}$	NA	NA	NA	5×10^{1}†

* Based on average release volume of 100 m^3 discharged three times per year. Measurements of the April 14, 1971 sample were not used to estimate the annual release.

** Analytical results for ^3H include both gas and H_2O, and for ^{14}C include both CO_2 and non-CO_2

† Calculated from ^{135}Xe measurements of September 16, 1970 and April 16, 1971 by Haddam Neck staff

Notes:

1. ± values indicate analytical error expressed at 2-sigma; < values are minimum detectable concentrations at 3-sigma counting error.
2. NA – not analyzed

Date	^{135}Xe concentration, µCi/cc
Sept. 16, 1970	1.6×10^{3}
March 15, 1971	not detectable
April 14, 1971	4.7×10^{4}
April 16, 1971	5.1×10^{4}

The amount of radioactivity in stored gas discharged annually to the atmosphere was calculated from the averages of the concentrations in the samples collected on July 27, 1970, September 16, 1970, March 15, 1971, and April 16, 1971, multiplied by the estimated discharge volume of 300 m^3 per year. The total is 160 Ci, mostly ^{133}Xe and ^{85}Kr. Compared to the discharges of stored gas reported by the station for the second half of 1970 (see Appendix B.4), these ^{85}Kr values are higher, ^{135}Xe values are similar, and ^{133}Xe values are lower.

Gaseous ^{131}I was not observed in any stored gas sample (minimum detectable level: 3×10^{5} µCi/cc). Stack effluent measurements, however, show that discharge rates of gaseous ^{131}I increased whenever surge sphere waste was released (Section 3.3.12). Its average concentration in stored gas during discharge can be inferred by multiplying the average stack concentration increment (Table 3.12, fifth column) by the dilution factor for stored gas (12,000:1 when it is released at an average flow of 2.75×10^{3} cc/s and the stack discharge rate is 3.3×10^{7} cc/s). By this calculation, the concentration of gaseous ^{131}I in the surge sphere averaged 2.2×10^{4} µCi/cc, and the annual discharge was 6.4×10^{4} Ci (see Table 3.12).

During release of stored gas, most particulate radionuclides were also at higher concentrations in the stack, as shown in Section 3.3.11. Their concentrations in effluent from the surge sphere were very small; concentrations in the surge sphere may have been higher, but can not be computed because the degree of retention on the glass fiber filter is not known.

Annual discharges of radioactive gases estimated from surge sphere measurements (last column in Table 3.2) are consistent with values computed from concentrations in reactor coolant water (Section 3.3.1) and the amount of reactor coolant water treated in the boron recovery system (Section 2.1.5). Annual discharges would be as follows, based on the average of gas concentration values from Table 3.1, converted to concentrations in water, and a daily water volume of 5,500 kg (1.8×10^{6} kg/yr in 330 days):

Radionuclide	Concentration in reactor coolant water, µCi/g	Calculated daily amount, Ci	Calculated annual discharge, Ci
^3H (gas)	8.4×10^{7}	4.6×10^{4}	1.5×10^{3}
^{14}C(non-CO_2)	3.5×10^{5}	1.9×10^{4}	6.3×10^{2}
^{85}Kr	1.8×10^{2}	9.9×10^{2}	3.3×10^{1}
^{131m}Xe	2.1×10^{2}	1.2×10^{1}	1.1
^{133}Xe	1.4	7.7	1.8×10^{3}
^{135}Xe	1.8×10^{1}	9.9×10^{1}	1.7

The annual discharges of long-lived ^3H, ^{14}C, and ^{85}Kr are based on accumulation for 330 days. The discharges of shorter-lived ^{131m}Xe, ^{133}Xe, and ^{135}Xe are equilibrium amounts (the average daily amount multiplied by the half-life and divided by 0.693) released three times per year; the differences between the summed daily amounts and the calculated annual discharge—38, 2,400, and 330 Ci, respectively—are due to decay in the sphere.

The annual values calculated above include any radioactive gases discharged with liquid waste from the reactor system, and hence may exceed the totals for gaseous releases given in the last column of Table 3.2. The station has reported ^{133}Xe and ^{135}Xe discharges in liquid waste. The entire amount of ^{133}Xe in the period July-December 1970 was from test tanks—i.e., reactor-system liquid waste—according to the values given in Appendix B.4.

The releases of radionuclides from the gas processing systems estimated in the Environmental Statement(8) on the basis of a model station adjusted for operating parameters at Haddam Neck are considerably higher for ^{85}Kr and ^{133}Xe:

^{85m}Kr	13 Ci/yr	^{131m}Xe	40 Ci/yr
^{85}Kr	480	^{133m}Xe	3
^{88}Kr	8	^{133}Xe	545
		^{135}Xe	2

These values, however, are based on 0.25 percent leakage of fission products from fuel rather than the actual leakage of 0.02 percent (see Section 2.3.3) reported by the station and a holdup time in the gas storage tank of only 30 days.

3.3.3 *Radionuclides in vapor container air.* Samples of air from the vapor container during reactor operation showed the presence of ^{85}Kr, ^{133}Xe, and ^{133m}Xe, as well as ^{3}H and ^{14}C in various chemical forms (see Nov. 20, 1970, and March 16, 1971 samples in Table 3.3). Samples of condensed water vapor from the vapor container collected at the same time (see Table 3.3) showed some of the same long-lived radionuclides that were observed in reactor coolant water (see Table 2.1).

The ambient concentrations of ^{3}H in water from vapor container air is assumed to be the value computed from the value for condensed water vapor; that measured directly in air is believed to be low because non-radioactive steam is customarily injected into the air to reduce the ^{3}H concentration before personnel entry into the building. The two sets of values compare as follows:

Date	^{3}H in air (condensed water vapor)
Nov. 20	1.14×10^{-5} ml/cc x 2.2×10^{1} µCi/ml = 2.5×10^{-4} µCi/cc
Mar. 16	8.3×10^{-6} ml/cc x 1.2 µCi/ml = 9.5×10^{-6} µCi/cc

Date	^{3}H in air (direct)
Nov. 20	1.7 x 10[illegible] µCi/cc
Mar. 16	3.2 x 10[illegible] µCi/cc

The concentration of ^{3}H in the condensed water vapor was lower than in reactor coolant water (see Table 2.1) but higher than in secondary coolant water (Table 2.2).

In the air sample of May 7, 1971 (after the container had been open to the outside for 3 weeks), ambient concentrations of the two most abundant radioactive gases were, on the average, 1,000-fold lower than in the two samples collected earlier, during reactor operation. These two earlier samples were taken to represent vapor container air at the time of shutdown.

Annual releases from venting the vapor container atmosphere depend upon the number and duration of reactor shutdowns for major maintenance and refueling. Plant reports indicate that the building is exhausted an average of once per year each for maintenance and for refueling.(4) The amount of activity discharged annually was calculated as the average of the concentrations found on November 20, 1970 and March 16, 1971, times 2 shutdowns, times an assumed air volume of 31,500 m^{3} per shutdown. To this value was added the amount of radionuclides released during refueling after the container is initially purged of the accumulated radionuclides: the radionuclide concentrations on May 7, 1971, multiplied by 8.7×10^{6} m^{3} (for an exhaust rate of 142 m^{3}/min during a 43-day period from April 19 to June 1, 1971). The total releases of the two relatively abundant radionuclides were, thus:

	^{85}Kr, Ci	^{133}Xe, Ci
accumulated radionuclides discharged immediately after reactor shutdown	72	78
radionuclides discharged continuously during refueling	6	43
annual total	78	121

The annual releases of all gaseous radionuclides, estimated by this procedure, are shown in Table 3.3. Tritium appears to be discharged mostly during refueling, while the other radionuclides are discharged mostly at the time of shutdown.

For comparison, the amounts of radionuclides that would accumulate in the containment vessel and be discharged twice yearly were calculated. At the average concentrations in reactor coolant water of gaseous radionuclides (see Section 3.3.2) and radioiodine (see Table 2.1), and the assumed leakage rate into the containment of 110 kg/day for 330 days, the discharges would be:

Radionuclide	Average concentration in reactor coolant water, µCi/g	Calculated daily leakage, Ci	Calculated annual discharge, Ci
^{3}H (gas)	8.4×10^{-1}	9.2×10^{-2}	3.0×10^{1}
^{14}C (non-CO_2)	3.5×10^{-4}	3.8×10^{-5}	1.3×10^{-2}
^{85}Kr	1.8×10^{-2}	2.0×10^{-3}	6.6×10^{-1}
^{131m}Xe	2.1×10^{-2}	2.3×10^{-3}	1.5×10^{-2}
^{133}Xe	1.4	1.5×10^{-1}	2.3
^{135}Xe	1.8×10^{-1}	2.0×10^{-2}	2.2×10^{-2}
^{131}I	2.3×10^{-2}	2.5×10^{-4}	5.8×10^{-4}
^{133}I	5.2×10^{-2}	5.7×10^{-4}	1.4×10^{-4}

The annual discharge values were calculated as were those for the gas processing system in Section 3.3.2. In addition, the ^{131}I and ^{133}I values are based on an air-water partition factor of 0.1,(8) where that is the ratio of the iodine in air to the iodine in air plus water in the building.(10) The discharges based on samples of

Table 3.3

Radioactivity in Vapor Container Atmosphere

Radionuclide	In air, µCi/cc Nov. 20, 1970	March 16, 1971	May 7, 1971	In condensate, µCi/ml Nov. 20, 1970	March 16, 1971	Estimated annual release of gases, Ci
^{3}H (gas)	$2.7 \pm 0.3 \times 10^{-7}$	$1.6 \pm 0.3 \times 10^{-7}$	$< 3 \times 10^{-8}$	NA	NA	1.9×10^{3}
^{3}H (H_2O)	$1.7 \pm 0.5 \times 10^{-7}$	$3.2 \pm 0.2 \times 10^{-7}$	$1.6 \pm 0.2 \times 10^{-8}$	$2.2 \pm 0.1 \times 10^{-1}$	1.2 ± 0.1	1.6×10^{1}
^{14}C (non-CO_2)	$1.9 \pm 0.1 \times 10^{-6}$	$2.5 \pm 0.1 \times 10^{-6}$	$< 1 \times 10^{-8}$	$4.2 \pm 0.4 \times 10^{-7}$	$2.4 \pm 0.3 \times 10^{-7}$	1.4×10^{1}
^{14}C (CO_2)	$7.9 \pm 0.7 \times 10^{-6}$	NA	$4 \pm 2 \times 10^{-8}$	NA	NA	4.0×10^{3}
^{54}Mn	NA	NA	NA	$1.8 \pm 0.6 \times 10^{-7}$	$5 \pm 2 \times 10^{-7}$	--
^{58}Co	NA	NA	NA	$3.0 \pm 0.8 \times 10^{-7}$	$6 \pm 2 \times 10^{-7}$	--
^{60}Co	NA	NA	NA	$2 \pm 1 \times 10^{-7}$	$1.1 \pm 0.3 \times 10^{-6}$	--
^{85}Kr	$8.9 \pm 0.1 \times 10^{-4}$	$1.4 \pm 0.1 \times 10^{-3}$	$7.1 \pm 0.2 \times 10^{-7}$	NA	NA	7.8×10^{3}
^{89}Sr	NA	NA	NA	$< 1 \times 10^{-8}$	$< 3 \times 10^{-7}$	--
^{90}Sr	NA	NA	NA	$8 \pm 3. \times 10^{-9}$	$3.8 \pm 0.1 \times 10^{-7}$	--
^{131}I	$< 2 \times 10^{-7}$	NA	NA	$1.8 \pm 0.1 \times 10^{-5}$	$1.6 \pm 0.4 \times 10^{-5}$	--
^{133m}Xe	$3.5 \pm 0.9 \times 10^{-4}$	$5 \pm 1 \times 10^{-4}$	$< 1 \times 10^{-4}$	NA	NA	3×10^{-1}
^{133}Xe	$7.7 \pm 0.1 \times 10^{-4}$	$1.7 \pm 0.1 \times 10^{-3}$	$4.9 \pm 0.1 \times 10^{-4}$	NA	NA	1.2×10^{2}
^{134}Cs	NA	NA	NA	$1.6 \pm 0.5 \times 10^{-7}$	$6 \pm 2 \times 10^{-7}$	--
^{137}Cs	NA	NA	NA	$2.6 \pm 0.6 \times 10^{-7}$	$4 \pm 3 \times 10^{-7}$	---

Notes:

1. ± values indicate analytical error expressed at 2-sigma; < values are minimum detectable concentrations at 3-sigma.
2. NA – not analyzed
3. Ambient water vapor concentration: 11.4 g/m³ on Nov. 20
   8.3 g/m³ on Mar. 16

containment air are higher than these calculated values by approximately two orders of magnitude. Possibly some of the calculational assumptions are erroneous; for example, the leakage rate of these gases may exceed that of the water, as suggested in Section 3.3.1.

The annual discharge of radionuclides by purging the containment, assumed to occur four times yearly, was estimated to be as follows in the Environmental Statement:(8)

^{85}Kr	15 Ci/yr	^{133}Xe	130 Ci/yr
^{131m}Xe	2	^{131}I	0.11
^{133m}Xe	1	^{133}I	0.02

These estimated values agree with the discharge values in Table 3.3 for ^{133}Xe, are higher for ^{133m}Xe, and lower for ^{85}Kr.

3.3.4 *Radionuclides in primary auxiliary building air.* The long-lived gases ^{3}H (in water vapor), ^{85}Kr, and ^{133}Xe were observed in a single sample of air from the primary auxiliary building (see Table 3.4). These airborne radionuclides presumably leaked from reactor coolant water and from liquid wastes.

Emission rates to the stack plenum, computed for the ventilation rate given in the note to Table 3.4, yield an annual discharge of approximately 700 Ci, almost entirely ^{133}Xe. The ratio of 71:1 for short-lived ^{133}Xe to long-lived ^{85}Kr in the sample of building air is similar to that in the reactor coolant gas sample of February 9, 1971 (see Table 3.1).

The amounts of gaseous radionuclides accompanying reactor coolant water leaking into the building at the rate of 75 kg/day (see Section 3.3.1) would be:

Radionuclide	Calculated annual leakage, Ci/year	Radionuclide	Calculated annual leakage, Ci/year
^{3}H (gas)	2.1×10^{-3}	^{133m}Xe	6.1×10^{-1}
^{14}C(non-CO$_2$)	8.6×10^{-4}	^{133}Xe	3.5×10^{1}
^{85}Kr	4.5×10^{-1}	^{135}Xe	4.4

The above are the discharges into the vapor container associated with 75 x 330 = 2.5 x 10^{4} kg of water per year at the concentrations computed in Section 3.3.2. If the unmeasured noble gas radionuclides listed in Section 3.3.1 are at the concentrations computed in Appendix C.2, then their amounts in 2.5 x 10^{4} kg of water would total 0.8 Ci/yr. The values computed above for ^{85}Kr and ^{133}Xe, however, are 20-fold less than the annual releases estimated in Table 3.4 from the measurement of ventilating air; hence the other computed values may also be low.

The amount of water vapor ^{3}H in building air would be 8 x 10^{3} µCi/day, if 2.6 kg of leaking reactor coolant water flash daily into the building atmosphere (see Section 2.1.5) at a concentration of 3 µCi/ml (see Appendix B.1 for February, 1971). This equals 3 Ci/yr, which is within a factor of 2 of the estimated value in Table 3.4.

The amounts of radioiodine discharged with building air per year would be 3.1 x 10^{-3} Ci ^{131}I, 7.0 x 10^{-3} Ci ^{133}I, and 5.4 x 10^{-3} Ci ^{135}I, if 2.5 x 10^{4} kg of reactor coolant water leak each year at the averages of the radioiodine concentrations given in Table 2.1. This is based on an iodine partition factor of 0.005(8) that takes into account leakage of both hot and cold reactor coolant water into the building.(10)

The annual release of radioactive gases from the auxiliary building was estimated as follows in the Environmental Statement:(8)

^{85}Kr	7 Ci/yr
^{131m}Xe	5
^{133m}Xe	9
^{133}Xe	760
^{135}Xe	13
^{131}I	0.02
^{133}I	0.04
Sum of short-lived radiokrypton	16
Sum of short-lived radioxenon	3

Table 3.4

Gaseous Radioactivity in Primary Auxiliary Building Atmosphere, Sample of February 9, 1971

Radionuclide	Concentration, µCi/cc	Emission rate, µCi/s	Estimated annual release,* Ci
^{3}H (gas)	$<3 \times 10^{-7}$	$<3 \times 10^{-2}$	$<9 \times 10^{-1}$
^{3}H (H$_2$O)	$1.6 \pm 0.2 \times 10^{-4}$	1.5×10^{1}	4.3
^{14}C	$<6 \times 10^{-7}$	$<6 \times 10^{-2}$	<2
^{85}Kr	$3.4 \pm 1.0 \times 10^{-6}$	3.2×10^{-1}	9.1
^{133}Xe	$2.4 \pm 0.1 \times 10^{-4}$	2.3×10^{1}	6.5×10^{2}

Notes:

1. Emission rates calculated for discharge rate of 9.5 x 10^{4} cc/s. Annual release based on 330 operating days per year.
2. ± values indicate analytical error expressed at 2-sigma; < values are minimum detectable levels at 3-sigma.

These estimated values of ^{85}Kr and ^{133}Xe agree with the annual releases computed in Table 3.4.

3.3.5 *Radionuclides discharged from secondary coolant system at main condenser steam jet air ejector.* All radionuclides observed in reactor coolant gas except ^{41}Ar were measured in gas discharged from the SJAE, as shown in Table 3.5. The concentrations of noble gases were approximately 10,000-fold lower than in the reactor coolant (Table 3.1) on November 20, 1970, and February 9, 1971. Radionuclide concentrations in the 8 secondary coolant samples varied within a factor of ten due to changes in such factors as reactor coolant concentrations and the leakage rate from reactor to secondary coolant.

Flow rates in the air ejector exhaust line at times of sampling are given in the last line of Table 3.5 for calculating radionuclide release rates during sampling. Annual releases of each radionuclide were estimated by averaging these release rates and then multiplying the average by an operating period of 330 days per year. Of the annual discharge of 1,200 Ci, 90 percent is contributed by ^{133}Xe.

The discharge data for off-gas at the SJAE reported by the station for July-December 1970 (see Appendix B.4) are of the same magnitude as the values for ^{133}Xe and ^{135}Xe in Table 3.5. In addition, 0.4 Ci ^{41}Ar was reported discharged in one month.

Concentrations of tritium in off-gas water vapor were computed from the tritium concentrations in steam-generator water shown in Table 2.2. The temperature of off-gas at the air ejector is given as 43° C (110° F),(9) at which saturated air contains 60 g of water vapor per cubic meter. The tritium concentrations at this water content would be:

Sampling date	Calculated ^{3}H concentration in off-gas water vapor, μCi/cc
July 24, 1970	8.5×10^{-7}
Sept.15, 1970	2.0×10^{-7}
Nov. 20, 1970	7.2×10^{-7}
Mar. 15, 1971	1.3×10^{-6}
Apr. 14, 1971	1.3×10^{-6}

The single measured value on these dates—that of March 15—is in agreement with these values (see Table 3.5). The annual release of tritiated water vapor computed from these concentrations is 1.3×10^{-1} Ci/year, compared to 6.1×10^{-2} Ci/year on the basis of the three measured values in Table 3.5.

The discharges of short-lived radionuclides of krypton and xenon and of ^{131m}Xe were estimated by assuming that these radionuclides were in the same amounts relative to ^{133}Xe as computed for the reactor coolant water in Appendix C.2:

(^{133}Xe	1100	Ci/yr from Table 3.5, last column)
^{85m}Kr	1.6	
^{87}Kr	3.2	
^{131m}Xe	15.6	
^{135m}Xe	0.4	
^{137}Xe	0.6	
^{138}Xe	2.1	
Total	23.5	Ci/yr

Because of radioactive decay in transit, the amounts of short-lived gases may be less. The measured amounts of the krypton and xenon isotopes given in Table 3.5 agree with the amounts computed as above within a factor of 2 or better, except that ^{88}Kr is 4-fold lower. The amount of ^{41}Ar was below the value of 2 Ci/yr corresponding to the detection limit; if the concentration relative to the other short-lived gases were similar to the values in Table 3.1, approximately 0.7 Ci of ^{41}Ar would be discharged annually.

The discharge of ^{131}I was calculated from the average concentration in steam generator water of 3.2×10^{-7} Ci/kg (from Table 2.2), the steam flow rate of 3.5×10^{6} kg/h(see Figure 2.1), and the air/water partition factor for iodine in the model plant(10) of 5×10^{-6}. This factor is composed of partition factors of 0.01 at the steam generators and 0.0005 at the SJAE. The ^{131}I release, therefore, would be $3.2 \times 10^{-7} \times 3.5 \times 10^{6} \times 5 \times 10^{-6} = 5.6 \times 10^{-6}$ Ci/h,or 4.4×10^{-2} Ci/yr for 330 days of operation. The corresponding discharge values for ^{133}I and ^{135}I, at respective average concentrations in steam generator water of 3.3×10^{-7} and 1.4×10^{-7} Ci/kg (from Table 2.2), are 4.5×10^{-2} and 1.9×10^{-2} Ci/yr.

The following annual discharges of radioactive gases were estimated in the Environmental Statement at the SJAE of the main condenser in the secondary system:(8)

^{83m}Kr	1	Ci/yr	^{133}Xe	770	Ci/yr
^{85m}Kr	5		^{135m}Xe	1	
^{85}Kr	8		^{135}Xe	14	
^{87}Kr	2		^{138}Xe	2	
^{88}Kr	8		^{131}I	0.03	
^{131m}Xe	5		^{133}I	0.03	
^{133m}Xe	9				

For the seven radioactive noble gases listed in Table 3.5, these estimates are approximately two-fold lower than the annual releases based on measured values. This difference is not excessive in view of the uncertainties of the estimate and the variability of measured values.

The main difficulty in relating the measured concentrations of radioactive gases to their turnover in the secondary system is that the discharge rate at the SJAE exceeds the inleakage rate at the steam generators. On an annual basis, the estimated releases of the noble gases in Table 3.5 divided by the computed concentrations in reactor coolant given in Appendix

Table 3.5

Radioactivity Contents of Discharge from Main Condenser Air Ejector in Secondary Coolant System

Radionuclide	Concentration, µCi/cc				
	July 24, 1970	Sept. 16, 1970	Nov. 20, 1970	Feb. 9, 1971	Mar. 15, 1971
^{3}H(gas)	3.5±0.6 x 10^{7}	<1 x 10^{7}	<4 x 10^{8}	3 ±1 x 10^{8}	1.0±0.1 x 10^{7}
^{3}H(H_2O)		NA	NA	6 ±1 x 10^{8}	1.1±0.1 x 10^{6}
^{14}C (non-CO_2)	2.6±0.4 x 10^{7}	2.1±0.3 x 10^{7}	1.1±0.3 x 10^{7}	2.1±0.2 x 10^{7}	5.3±0.2 x 10^{7}
^{14}C (CO_2)	NA	NA	NA	NA	[illegible] ±1 x 10^{8}
^{41}Ar	NA	NA	NA	<2 x 10^{6}	NA
^{85m}Kr	NA	NA	NA	<2 x 10^{6}	NA
^{85}Kr	6 ±2 x 10^{5}	1.1±0.1 x 10^{4}	1.1±0.1 x 10^{5}	4.4±0.6 x 10^{5}	1.1±0.1 x 10^{4}
^{87}Kr	NA	NA	NA	<4 x 10^{5}	NA
^{88}Kr	NA	NA	NA	<5 x 10^{6}	NA
^{133m}Xe	NA	1.1±0.1 x 10^{4}	3.4±0.2 x 10^{5}	1.2±0.1 x 10^{4}	1.9±0.2 x 10^{4}
^{133}Xe	6.2±0.1 x 10^{3}	7.7±0.1 x 10^{3}	2.4±0.1 x 10^{3}	5.5±0.1 x 10^{3}	1.3±0.1 x 10^{2}
^{135}Xe	NA	3.0±0.2 x 10^{4}	1.3±0.2 x 10^{4}	4.9±0.1 x 10^{4}	2.1±0.8 x 10^{4}
Flow rate, cc/sec	6170	6700	6170	8030	3780

Radionuclide	Concentration, µCi/cc			Estimated annual release,* Ci
	Mar. 16, 1971	Apr. 14, 1971	Apr. 16, 1971	
^{3}H(gas)	1.3±0.1 x 10^{7}	NA	NA	2.0 x 10^{3}
^{3}H(H_2O)	5 ±3 x 10^{7}	NA	NA	6.1 x 10^{3}
^{14}C (non-CO_2)	6.2±0.2 x 10^{7}	NA	NA	4.6 x 10^{2}
^{14}C (CO_2)	NA	NA	NA	5 x 10^{3}
^{41}Ar	<4 x 10^{7}	NA	NA	<2
^{85m}Kr	4.2±0.3 x 10^{6}	NA	NA	6.6
^{85}Kr	1.5±0.1 x 10^{4}	8 ±2 x 10^{6}	7 ±1 x 10^{6}	1.2 x 10^{1}
^{87}Kr	8 ±1 x 10^{6}	NA	NA	1 x 10^{1}
^{88}Kr	1.1±0.1 x 10^{4}	NA	NA	1.7 x 10^{1}
^{133m}Xe	2.5±0.2 x 10^{4}	1.4±0.1 x 10^{4}	1.1±0.1 x 10^{4}	1.8 x 10^{1}
^{133}Xe	1.5±0.1 x 10^{3}	7.3±0.1 x 10^{3}	5.2±0.1 x 10^{3}	1.1 x 10^{3}
^{135}Xe	1.7±0.1 x 10^{4}	NA	9.6±0.9 x 10^{4}	5.8 x 10^{1}
Flow rate, cc/sec	3530	4250	4250	—

*Based on the average of the emission rates in µCi/s multiplied by 330 days (2.85 x 10^{7} s) of reactor operation per year.

Notes:

1. ± values indicate analytical error expressed at 2σ;
   < values are minimum detectable concentrations at 3σ counting error.
2. NA – not analyzed.

C.2 require an average inleakage rate of 3 x 10^5 kg of reactor coolant water. This rate (900 kg/day) is several times higher than the reported value of 75 to 150 kg/day.(8) On the two occasions when gas samples were collected both from reactor coolant and at the SJAE, the inleakage and discharge rates compare as follows:

	Nov. 20, 1970		Feb. 9, 1971	
Radionuclide	Leakage into secondary system, µCi/s	Discharge at SJAE, µCi/s	Leakage into secondary system, µCi/s	Discharge at SJAE, µCi/s
^{3}H (gas)	4.5x10^{-7}	<4.5x10^{-7}	1.7x10^{-4}	2.4x10^{-4}
^{14}C(non-CO_2)	7.6x10^{-5}	6.8x10^{-4}	1.4x10^{-5}	1.7x10^{-1}
^{85m}Kr	---	---	5.9x10^{-2}	<2. x10^{-1}
^{85}Kr	1.7x10^{-2}	6.8x10^{-2}	3.1x10^{-2}	3.5x10^{-1}
^{87}Kr	---	---	7.6x10^{-2}	<3. x10^{-1}
^{88}Kr	---	---	1.1x10^{-1}	<4. x10^{-1}
^{133m}Xe	5.7x10^{-2}	2.1x10^{-1}	3.2x10^{-2}	9.6x10^{-1}
^{133}Xe	1.4	1.5x10^{1}	2.3	4.4x10^{1}
^{135}Xe	7.6x10^{-3}	8.0x10^{-1}	3.8x10^{-1}	3.9

The gas inleakage rate was calculated at the gas concentrations of Table 3.1, for 35 cc gas per kg reactor coolant water and an average water inleakage rate of 110 kg reactor coolant per day; the discharge at the SJAE is at the concentrations and gas flow rates given in Table 3.5 for the two sampling dates. Even a 2-fold higher rate of water inleakage (see computed values in Section 2.3.5) would not bring the radionuclide flow rates into balance. The apparently higher discharges could be due to nonrepresentative samples or a greater leakage rate for gases than for water.

3.3.6 *Radionuclides in turbine building air.* The only radionuclides observed in a single sample of air collected in the turbine building were ^{3}H (in water vapor) and ^{85}Kr at the concentrations given in Table 3.6. During 330 days of operation per year at an air turnover of 4.7 x 10^7 cc/s (100,000 cfm),(8) 150 Ci of ^{3}H and 43 Ci ^{85}Kr would be discharged per year.

The annual discharges calculated for these two radionuclides at an assumed steam leakage rate from the secondary system of 9,300 kg/day (see Section 3.1.1) are considerably lower. If 35 percent of the leaking steam remains as vapor (see Section 2.1.5), the ^{3}H discharge at an average ^{3}H concentration in steam of 1.4 x 10^{-5} Ci/kg (from Table 2.2) would be 15 Ci/yr. For ^{85}Kr and all other noble gases, the annual discharge can be taken to be 1 x 10^{-4} of the values at the SJAE (see Table 3.5), that being the ratio of the steam leakage rate to the steam flow of 3.5 x 10^6 kg/h in the secondary system. Hence, the ^{85}Kr discharge would be 1.2 x 10^{-3} Ci/yr. The ten-fold higher measured value of ^{3}H may be due to incomplete mixing of air at the sampling point, steam leakage greater than estimated, or both. The 36,000-fold higher measured value of ^{85}Kr suggests that the radioactive gas was from a source other than secondary system leakage. Additional samples should be collected to check these values.

The annual discharges of ^{131}I, ^{133}I, and ^{135}I, computed from their average concentrations in steam generator water (see Section 3.3.5), a steam/water partition factor of 1.0 relative to the steam—i.e., 0.01

Table 3.6

Gaseous Radioactivity in Turbine Hall Atmosphere, Sample of April 15, 1971

Radionuclide	Concentration, µCi/cc	Release rate, µCi/s	Estimated annual release, Ci
^{3}H (gas)	< 3 x 10^{-9}	<2 x 10^{-1}	<4
^{3}H (H_2O)	1.1±0.1 x 10^{-7}	5.2	1.5 x 10^2
^{14}C	< 8 x 10^{-9}	<4 x 10^{-1}	<1 x 10^1
^{85}Kr	3.2±0.8 x 10^{-8}	1.5	4.3 x 10^1
^{133}Xe	< 3 x 10^{-7}	<1 x 10^1	<4 x 10^2

Notes:

1. Release rates calculated for a building discharge rate of 4.7 x 10^7 cc/s. Annual release computed for 330 operating days per year.
2. ± values indicate analytical error expressed at 2-sigma; < values are minimum detectable levels at 3-sigma.

relative to steam-generator water—and an annual leakage rate of 3.1 x 10^6 kg, would be 0.010, 0.010, and 0.004 Ci, respectively. The estimated annual discharges scaled down by a factor of two from the 2-fold larger model plant(10) are similar: 0.012 Ci for ^{131}I, and 0.008 Ci for ^{133}I. According to the model, the release of each noble gas radioisotope is less than 0.5 Ci/yr.

3.3.7 *Radionuclides discharged from air ejector at turbine gland seal condenser.* Samples of this effluent were not available, but should be collected to check estimated radionuclide discharges. Calculations suggest that ^{3}H in water vapor may be the radionuclide at highest concentration. At a temperature of 52° C (125° F)(9), for which the saturation concentration of water vapor is 90 g/m^3, and an average ^{3}H concentration of 1.4 x 10^{-2} µCi/g in steam-generator water (from Table 2.2), the effluent gas would have a ^{3}H concentration of 1.3 µCi/m^3. For each 4,100 m^3 of gas flowing per day (100 cfm), the annual (330-day) discharge of ^{3}H would be 1.8 Ci. The actual flow rate of noncondensable gases was not known, but is believed to be of this magnitude.

The discharge of radioactive noble gases is inferred to be 1 x 10^{-3} of that at the SJAE (see Table 3.5), in that 0.1 percent of the steam flows through the gland seal system.(8) This would include 1 Ci ^{133}Xe per year, and considerably lesser amounts of the other noble gas radionuclides.

The ^{131}I discharge was calculated from the average concentration in steam generator water of 3.2 x 10^{-7} Ci/kg (see Section 3.3.5), the steam flow rate in the gland seal system of 3.5 x 10^3 kg/h,and a partition factor for iodine of 1 x 10^{-5} relative to steam-generator water (a steam/water ratio of 0.01 in the steam generators, 0.01 in the gland seal condenser, and 0.1 in the gland seal air ejector).(8,10) In 330 days of operation, this yields an annual ^{131}I discharge of 9 x 10^{-5} Ci. An equal amount of ^{133}I and less ^{135}I (see Section 3.3.5) would accompany the ^{131}I.

3.3.8 *Radionuclides discharged at blowdown flash tank vent.* Samples from this source also were not available for analyzing the radioactive effluent; however, it is discharged through the vent stack, hence the samples described in Section 3.3.10 through 3.3.12 include this effluent. Calculations suggest that ^{3}H in water vapor is the radionuclide at highest concentration, and that most of the airborne radioiodine is released from the station via this pathway.

The discharge of ^{3}H in water vapor was computed from the annual amount of 2.2 x 10^6 kg blowdown steam estimated in Section 2.1.5, at an average ^{3}H concentration (from Table 2.2) of 1.4 x 10^{-5} Ci/kg. This yields an annual ^{3}H discharge of 31 Ci.

The discharge of ^{131}I in the gaseous effluent was computed from the total annual blowdown (water plus steam) of 6.3 x 10^6 kg (see Section 2.1.5), an average ^{131}I concentration of 3.2 x 10^{-7} Ci/kg (see Section 3.3.5), and a vapor/water partition factor of 0.05 for iodine in the flash tank.(8) Accordingly, the annual ^{131}I discharge would be 0.04 Ci. Discharges of ^{133}I and ^{135}I, at the concentrations given in Section 3.3.5, would be 0.04 and 0.018Ci/yr, respectively. The estimated values for ^{131}I and ^{133}I in the Environmental Report are 0.20 and 0.18 Ci/yr, respectively.(8)

The concentrations of the radioactive noble gases in blowdown steam and water are not known. If it is assumed for an upper limit that their concentrations are the same as in turbine steam, and that all of the noble gases will accompany the flashing steam, the ratio of the discharge rate of radioactive noble gases at the blowdown vent to that at the SJAE would equal the ratio of the blowdown rate to the steam flow rate. This is approximately 790 kg/h divided by 3.5 x 10^6 kg/h = 2 x 10^{-4}. Relative to the values in Table 3.5, the sum of the radioactive noble gases amounts to less than 1 Ci/yr. The estimates for the model plant, adjusted for the two-fold smaller size of the Haddam Neck station, are less than 0.5 Ci/yr for each radioactive noble gas.(10)

3.3.9 *Radionuclides in fuel building air.* Only ^{3}H in water vapor, ^{14}C, and ^{85}Kr were observed in a single sample of ventilating air collected in the fuel building. The annual releases shown in Table 3.7 were computed on the basis of these measurements and a ventilation rate of 70 m^3/min for 365 days. These long-lived radionuclides would be expected to be associated with fuel stored more than 200 days since the previous refueling; short-lived radionuclides would appear during and immediately after refueling. If the moisture content of the ventilating air was approximately 1 x 10^{-5} g/cc, the measurement in air would reflect an ^{3}H concentration of 6.1 x 10^{-8} µCi/cc divided by 1 x 10^{-5} g/cc = 6.1 x 10^{-3} µCi/g water in the fuel pool.

3.3.10 *Radioactive gases discharged through the vent stack.* Radioactive gases were measured in stack samples on three occasions during routine continuous discharges and twice when surge sphere gases were also being released. The measured concentrations in Table 3.8 and the calculated release rates in Table 3.9 indicate the presence of ^{3}H, ^{85}Kr, and ^{133}Xe during continuous discharges, and larger amounts of the same radionuclides, as well as some ^{14}C, when gas from the surge sphere was added. The largest fraction of the annual discharge of ^{3}H and ^{133}Xe was due to continuous sources.

Table 3.7

Gaseous Radioactivity in Fuel Building Atmosphere, Sample of February 9, 1971

Radionuclide	Concentration, µCi/cc	Emission rate, µCi/s	Estimated annual release, Ci
^{3}H (gas)	$< 2 \times 10^{-6}$	$< 2 \times 10^{-2}$	$< 6 \times 10^{-1}$
^{3}H (H_2O)	$6.1 \pm 0.2 \times 10^{-6}$	7.3×10^{-2}	2.3
^{14}C	$9 \pm 2 \times 10^{-7}$	1.1×10^{-2}	3×10^{-1}
^{85}Kr	$2 \pm 1 \times 10^{-6}$	2.4×10^{-2}	8×10^{-1}
^{133}Xe	$< 3 \times 10^{-7}$	$< 4 \times 10^{-1}$	< 12

Notes:

1. Emission rates calculated for discharge rate of 1.2×10^{4} cc/s. Annual release based on 365 days per year.
2. ± values indicate analytical error expressed at 2-sigma; < values are minimum detectable levels at 3-sigma.

Table 3.8

Radionuclide Concentrations in Primary Vent Stack Effluents, µCi/cc

Radionuclide	Sept. 15, 1970	Sept. 16, 1970*	Mar. 16, 1971	Apr. 14, 1971	Apr. 16, 1971*
^{3}H (gas)	$1.3 \pm 0.8 \times 10^{-8}$†	$2.5 \pm 0.9 \times 10^{-8}$†	$< 3 \times 10^{-9}$	$< 3 \times 10^{-9}$	$1.5 \pm 0.3 \times 10^{-8}$
^{3}H (H_2O)			$4.3 \pm 0.1 \times 10^{-7}$	$3.9 \pm 0.4 \times 10^{-8}$	$1.1 \pm 0.2 \times 10^{-8}$
^{14}C (non-CO_2)	$< 3 \times 10^{-9}$†	$1.7 \pm 0.2 \times 10^{-8}$†		$< 3 \times 10^{-9}$	$7.0 \pm 0.4 \times 10^{-8}$
^{14}C (CO_2)			$< 3 \times 10^{-9}$	$< 4 \times 10^{-9}$	$< 2 \times 10^{-9}$
^{85}Kr	$7 \pm 1 \times 10^{-8}$	$9.0 \pm 0.1 \times 10^{-6}$	$4.2 \pm 0.7 \times 10^{-8}$	$4 \pm 1. \times 10^{-8}$	$8.7 \pm 0.1 \times 10^{-6}$
^{133}Xe	$3.0 \pm 0.1 \times 10^{-6}$	$6.2 \pm 0.1 \times 10^{-5}$	$2.3 \pm 0.3 \times 10^{-6}$	$3.5 \pm 0.4 \times 10^{-6}$	$5.1 \pm 0.2 \times 10^{-5}$

*Samples were obtained during release of waste surge sphere gas when stack flow rate was 33.3 m^3/s. Flow rate for other samples was 16.6 m^3/s.

†Gaseous ^{3}H and ^{14}C measurements of Sept. 15 and 16, 1970, samples include all forms.

Note:

± values indicate analytical error expressed at 2-sigma; < values are minimum detectable concentrations at 3-sigma counting error.

Table 3.9

Average and Annual Estimated Radioactivity Releases from the Primary Vent Stack

	Average release rate, µCi/s		Estimated annual release, Ci	
Radionuclide	Continuous	Surge sphere plus continuous	Continuous	Surge sphere
^{3}H	0.43	0.85	12	0.05
^{14}C	$<$0.1	1.4	$<$ 3	0.17
^{85}Kr	0.9	290	26	35
^{133}Xe	53	1,900	1,500	220

Notes:

1. Continuous values are averages of results from Table 3.8 for Sept. 15, 1970, and April 14, 1971; surge sphere plus continuous values are averages of results for Sept. 16, 1970, and April 16, 1971; the values for ^{3}H in gas and H_2O have been combined, as are the values for ^{14}C in CO_2 and non-CO_2 gas.
2. Average release rates are calculated from average of concentrations for continuous or stored gas releases (Table 3.8) times stack discharge rate of 1.67×10^7 cc/s for continuous releases or 3.33×10^7 cc/s for stored gas releases.
3. Estimated annual releases for continuous discharge are given by multiplying average release rate by 330 days (2.85×10^7 s) of reactor operation. Annual stored gas releases are given by subtracting the continuous release rate from the release rate during discharge of stored gas and multiplying this difference by 33.3 hrs (1.20×10^5 s), the estimated time required to release 300 m^3 of waste per year at average flow rate of 2500 cc/s.

The averages of measured stack values were used to check some of the previously described measurements at the sources of the effluent gases. For releases from the surge sphere, annual discharges compare as follows:

	Estimated annual discharge from surge sphere, Ci	
Radionuclide	Surge sphere samples (Table 3.2)	Stack samples (Table 3.9)
^{3}H	0.007	0.05
^{14}C	0.032	0.17
^{85}Kr	29	35
^{133}Xe	130	220

For continuous releases, the four contributing sources (see Figure 3.1) were summed: (1) SJAE (Table 3.5); (2) primary auxiliary building air (PAB, Table 3.4); (3) fuel building air (FB, Table 3.7); and (4) blowdown flash tank vent (BFT, Section 3.3.8). These compare to the continuous discharge values from stack samples in Table 3.9 as follows:

Radio-	Estimated annual continuous discharge, Ci					
nuclide	SJAE	PAB	FB	BFT*	Sum	Stack samples
^{3}H	0.081	4.3	2.3	(31)	7 to 38	12
^{14}C	0.051	$<$2	0.3	nc	0.35 to $<$2.4	$<$3
^{85}Kr	12	9.1	0.8	(0.01)	22	26
^{133}Xe	1,100	650	12	(1)	1,800	1,500

*Values were calculated, not measured; nc: not calculated, but believed to be smaller than the SJAE value.

The values are comparable except for the 5-to-7 times higher discharge values of ^{3}H and ^{14}C from the surge sphere on the basis of the stack samples. Considerable uncertainty is introduced into the comparison by such factors as sample collection at different times; possibly unrepresentative samples, as of building air; use of approximate flow rates (e.g., daily averages); and fluctuating discharges (e.g., much of the blowdown occurs during a few hours each day).

Specific comparisons of radionuclide discharge rates measured in the stack with the total of those measured on the same day in the principal pathways to the stack are presented in Table 3.10. The pathways include off-gas from the secondary steam condenser air ejector (Section 3.3.5), ventilation air exhausted from the primary auxiliary and fuel buildings (Sections 3.3.4 and 3.3.9, respectively) and waste surge sphere discharge (Section 3.3.2). Radionuclide concentrations were converted to discharge rates for the air flow rates given in footnotes to the utilized tables.

Table 3.10

Comparison of Gaseous Radionuclide Release Rates Measured in Plant Pathways and Stack

Radionuclide	Sampling date	Plant Pathways, μCi/sec: Secondary steam condenser	Primary auxiliary bldg. exhaust*	Fuel building exhaust*	Waste surge sphere discharge	Total	Primary vent stack discharge, μCi/s	Ratio, pathway/stack
			Routine waste release					
^{3}H (H_2O)	Mar. 16, 1971	2×10^{-3}	1.5×10^{-1}	7.3×10^{-2}	---	2.2×10^{-1}	7.1	0.03
^{85}Kr	Sept. 15, 1970	7.4×10^{-1}†	3.2×10^{-1}	2.4×10^{-2}	---	1.1	1.2	0.9
	Mar. 16, 1971	5.3×10^{-1}	3.2×10^{-1}	2.4×10^{-2}	---	8.7×10^{-1}	7.0×10^{-1}	1.2
	Apr. 14, 1971	3×10^{-1}	3.2×10^{-1}	2.4×10^{-2}	---	6.4×10^{-1}	7×10^{-1}	0.9
^{133}Xe	Sept. 15, 1970	5.2×10^{1}†	2.3×10^{1}	$< 3 \times 10^{-1}$	---	7.5×10^{1}	5.1×10^{1}	1.5
	Mar. 16, 1971	5.3×10^{1}	2.3×10^{1}	$< 3 \times 10^{-1}$	---	7.6×10^{1}	3.9×10^{1}	2.0
	Apr. 14, 1971	3.1×10^{1}	2.3×10^{1}	$< 3 \times 10^{-1}$	---	5.4×10^{1}	5.7×10^{1}	1.0
			Stored gas releases plus routine waste releases					
^{85}Kr	Sept. 16, 1970	7.4×10^{-1}	3.2×10^{-1}	2.4×10^{-2}	1.8×10^{2}	1.8×10^{2}	3.0×10^{2}	0.6
^{133}Xe	Sept. 16, 1970	5.2×10^{1}	2.3×10^{1}	$< 3 \times 10^{-1}$	1.8×10^{3}	1.9×10^{3}	2.1×10^{3}	0.9
	Apr. 16, 1971	2.2×10^{1}	2.3×10^{1}	$< 3 \times 10^{-1}$	8.3×10^{2}	8.8×10^{2}	1.7×10^{3}	0.5

* Samples of building ventilation exhaust air collected on Feb. 9, 1971.

† Samples of off-gas from secondary steam condenser air ejectors collected on Sept. 16, 1970.

Most results agree within a factor of two as shown by the ratios of pathway and stack values given in the last column of Table 3.10. The release rate of ^{3}H in water vapor was 30 times higher in the stack than in the summed pathways on March 16, 1971, but that sum does not include flashed steam from steam generator blowdown. This was computed to carry ^{3}H at an average rate of approximately 1 μCi/s (see Section 3.3.8). The higher ^{133}Xe stack effluent rate on April 16, 1971, may have arisen from operations for a planned shutdown later in the day.

3.3.11 *Radioactive particles discharged through the vent stack.* Airborne particles collected in the stack sampler contained the 9 long-lived activation and fission products listed in Table 3.11. Iron-55 usually was the most abundant radionuclide; only ^{55}Fe, ^{60}Co, and ^{90}Sr were detected in all samples. Radionuclide concentrations fluctuated among samples by approximately two orders of magnitude. Iodine-131 was found on the charcoal samplers and, in two instances, on filters, as discussed in Section 3.3.12. The main continuous sources of the particulate radioactivity are unfiltered air from the primary auxiliary and fuel buildings, and the unfiltered gases from the SJAE and the blowdown flash tank.

Concentrations of particulate radionuclides in stack samples taken during the three releases of waste surge sphere gas were higher than during continuous releases, suggesting that some radioactive particles are discharged with surge sphere gas despite filtering. The average concentrations during continuous release and the incremental concentrations while the stored gas from the surge sphere was released are shown in Table 3.12. The amounts released annually from the surge sphere were about an order of magnitude lower than those released continuously.

The total discharge of all long-lived radioactive particles in stack effluents listed in Table 3.12 is 3×10^{-3} Ci per year, of which ^{55}Fe constitutes 40 percent. Haddam Neck staff reported short-lived 17.8-min ^{88}Rb as the major particulate stack emission, in that 0.2 Ci was released in 1971 in addition to 1.8×10^{-2} Ci of other radionuclides (see Appendix B.3). The longer-lived radionuclides in Table 3.11 were, therefore, only a small fraction of the particulate effluent.

3.3.12 *Iodine-131 discharged through the vent stack.* Iodine-131 was found in all charcoal samplers, indicating that it is being discharged continuously (see Tables 3.11 and 3.12). Sample concentrations varied by approximately two orders of magnitude. The highest measured continuous release rate was 4×10^{-3} μCi/s, on September 15 to 16, 1970. This value may be related to reactor startup on the previous day.

Iodine-131 stack emissions were predominantly gaseous, although radionuclides could be found on the filters that preceded the charcoal during two relatively high releases. On these two occasions, approximately 4 percent of ^{131}I was collected on the filter.

The higher emission rates during waste surge sphere releases than during routine continuous plant discharge indicate that ^{131}I was present in the waste surge sphere although never measurable directly (see Section 3.3.1), as shown by the following sets of samples:

Date	Discharge rate, μCi/s routine	 stored gas
Sept. 16, 1970	4.0×10^{-3}	2.2×10^{-2}
March 15, 1971	1.3×10^{-4}	1.7×10^{-4}
April 16, 1971	8.1×10^{-5}	3.3×10^{-4}

The charcoal samplers used with large volumes of stack effluent are more sensitive for detecting ^{131}I than the relatively small samples of surge sphere gas.

The annual ^{131}I release was 1.3×10^{-2} Ci from continuous sources, as shown in Table 3.12, estimated by the method described in Section 3.3.11. Iodine-133 is also emitted from the stack but, because of its short (20.9-hr) half-life, was not measured in samples collected during this study. The station reported a discharge of 1.2×10^{-2} Ci ^{131}I and 1×10^{-3} Ci ^{133}I in 1971 (see Appendix B.3).

The estimated continuous discharges of ^{131}I by the various pathways leading into the vent stack, except for the unknown but expectedly small contribution from the fuel building (see Sections 3.3.9), compare as follows:

Source	Estimated annual ^{131}I discharge, Ci/yr based on ^{131}I concentration in secondary coolant water	 from Environmental Statement(8)
Primary auxiliary bldg. (Section 3.3.4)	0.0015	0.02
SJAE (Section 3.3.5)	0.04	0.03
Blowdown flash tank (Section 3.3.8)	0.04	0.20
Total	0.08	0.25

Table 3.11

Particulate Radionuclide and Gaseous Iodine-131 Concentrations in Stack Effluent, μCi/m^3

Radionuclide	1970 July 27–28	July 28–29	July 29–31	July 31-Aug. 3	Aug. 3–4
Particles on filter					
313 -day ^{54}Mn	2 ± 1 x10	1.9 ± 0.2x10	4.0 ± 0.6x10	1.2 ± 0.1x10	1 ± 1 x10
2.7 -yr ^{55}Fe	1.6 ± 0.1x10	2.8 ± 0.1x10	2.8 ± 0.1x10	1.4 ± 0.1x10	2.7 ± 0.1x10
71.3 -day ^{58}Co	7.5 ± 0.3x10	1.9 ± 0.1x10	3.7 ± 0.3x10	1.1 ± 0.1x10	1.3 ± 0.1x10
5.26-yr ^{60}Co	1.6 ± 0.2x10	4.1 ± 0.2x10	6.9 ± 0.1x10	2.0 ± 0.1x10	2.7 ± 0.2x10
50.5 -day ^{89}Sr	< 8 x10	< 1 x10	< 5 x10	< 3 x10	< 9 x10
28.5 -yr ^{90}Sr	1.4 ± 0.4x10	3.0 ± 0.4x10	2.8 ± 0.4x10	1.0 ± 0.1x10	2.5 ± 0.4x10
8.06-day ^{131}I	< 2 x10	< 5 x10	< 4 x10	< 7 x10	< 3 x10
2.07-yr ^{134}Cs	< 2 x10	< 4 x10	< 9 x10	< 3 x10	< 3 x10
30 -yr ^{137}Cs	< 2 x10	< 5 x10	< 9 x10	< 3 x10	< 3 x10
Gaseous iodine on charcoal					
8.06-day ^{131}I†	1.4 ± 0.1x10	1.5 ± 0.1x10	1.4 ± 0.1x10	9.8 ± 0.1x10	1.2 ± 0.1x10
Sample volume, m^3	65	57	134	220	63

Radionuclide	1970 Sept. 15–16	Sept. 16*	1971 March 15*	March 15–16	April 14–15	April 16*
Particles on filter						
313 -day ^{54}Mn	7 ± 3 x10	6 ± 3 x10	7.0 ± 0.6x10	< 2 x10	< 9 x10	1.5 ± 0.3x10
2.7 -yr ^{55}Fe	5.4 ± 0.1x10	4.5 ± 0.4x10	8.3 ± 0.6x10	4.0 ± 0.8x10	5 ± 1 x10	1.9 ± 0.1x10
71.3 -day ^{58}Co	1.8 ± 0.2x10	1.4 ± 0.5x10	4.3 ± 0.3x10	1.7 ± 0.2x10	2.8 ± 0.2x10	1.5 ± 0.8x10
5.26-yr ^{60}Co	7 ± 1 x10	< 7 x10	1.3 ± 0.1x10	1.0 ± 0.2x10	1.4 ± 0.1x10	3.4 ± 0.9x10
50.5 -day ^{89}Sr	< 5 x10	< 4 x10	< 8 x10	< 4 x10	2.7 ± 0.5x10	< 2 x10
28.5 -yr ^{90}Sr	1.7 ± 0.4x10	1.7 ± 0.2x10	5 ± 2 x10	4.9 ± 0.6x10	1.3 ± 0.4x10	6 ± 3 x10
8.06-day ^{131}I	9.3 ± 0.1x10	3.0 ± 0.5x10	< 3 x10	< 5 x10	< 2 x10	< 7 x10
2.07-yr ^{134}Cs	2.0 ± 0.1x10	2.3 ± 0.5x10	4.1 ± 0.6x10	< 1 x10	< 7 x10	1.0 ± 0.5x10
30 -yr ^{137}Cs	2.9 ± 0.5x10	4.3 ± 0.5x10	8.7 ± 0.5x10	< 2 x10	< 8 x10	3.1 ± 0.7x10
Gaseous iodine on charcoal						
8.06-day ^{131}I†	2.4 ± 0.1x10	6.6 ± 0.1x10	5.1 ± 0.2x10	7.9 ± 0.6x10	4.9 ± 0.6x10	1.0 ± 0.4x10
Sample volume, m^3	55	11	18	56	75	9

* Samples obtained during release of waste surge sphere gas when stack flow rate was 33.3m^3/s.
For other samples, flow rate was 16.7 m^3/s.

† Additional values: gaseous ^{131}I June 1–2, 1971: 8.2 ± 0.2 pCi/m^3, 24 m^3 air volume
2–3, 1971: 1.3 ± 0.1 pCi/m^3, 34 m^3 air volume
3–4, 1971: 1.0 ± 0.1 pCi/m^3, 42 m^3 air volume

Note: < values are minimum detectable concentrations at 3 sigma counting error; ± values are 2 sigma analytical error.

Table 3.12

Summary of Stack Release Rates and Estimated Annual Releases of Particulate and Gaseous Iodine Radionuclides

Radionuclide	Continuous release: Avg. concentration,* µCi/m^3	Continuous release: Avg. flow rate, µCi/s	Continuous release: Estimated annual release,** Ci	Stored gas release: Avg. concentration above continuous release,* µCi/m^3	Stored gas release: Estimated annual release,** Ci
Particles on filter					
^{54}Mn	2.0×10^{-7}	3.4×10^{-6}	9.7×10^{-5}	3.6×10^{-6}	1.4×10^{-5}
^{55}Fe	3.0×10^{-6}	4.7×10^{-5}	1.3×10^{-3}	4.2×10^{-5}	1.7×10^{-4}
^{58}Co	1.5×10^{-6}	2.3×10^{-5}	6.6×10^{-4}	1.3×10^{-6}	5.2×10^{-6}
^{60}Co	3.3×10^{-7}	5.6×10^{-6}	1.6×10^{-4}	6.6×10^{-6}	2.6×10^{-5}
^{89}Sr	2.8×10^{-8}	4.6×10^{-7}	1.3×10^{-5}	$<2 \times 10^{-7}$	$<8 \times 10^{-7}$
^{90}Sr	2.0×10^{-8}	3.3×10^{-7}	9.4×10^{-6}	6.7×10^{-8}	2.7×10^{-7}
^{131}I	7.1×10^{-7}	1.2×10^{-5}	3.4×10^{-4}	7.2×10^{-6}	2.9×10^{-5}
^{134}Cs	2.5×10^{-7}	4.2×10^{-6}	1.2×10^{-4}	2.6×10^{-6}	1.0×10^{-5}
^{137}Cs	2.1×10^{-7}	5.8×10^{-6}	1.7×10^{-4}	5.9×10^{-6}	2.4×10^{-5}
Gaseous iodine on charcoal					
^{131}I	2.8×10^{-5}	4.6×10^{-4}	1.3×10^{-2}	1.7×10^{-4}	6.8×10^{-4}

* Values below minimum detectable concentration levels in Table 3.11 were taken to be zero for averaging. All concentrations were weighted by sample volume.

**Estimated annual releases for continuous discharge are given by multiplying average release rate by 330 days (2.85×10^7 s) of reactor operation. Annual stored gas releases are given by subtracting the continuous release rate from the release rate during discharge of stored gas and multiplying this difference by 33.3 hrs (1.20×10^5 s), the estimated time required to release 330 m^3 of waste per year at average flow rate of 2500 cc/s.

The far lower measured value suggests either that ^{131}I discharge from the blowdown flash tank is grossly over-estimated, or that the charcoal sampler was not effective. The latter situation could be due to the chemical form of the iodine or to interferences, as by steam from the relatively large nightly blowdown (see Section 2.1.5).

3.3.13 *Estimated annual radionuclide discharges.* The measured effluent values discussed in the preceding parts of Section 3.3 provide the radioactivity source terms for planning environmental measurements. The total discharged radioactivity and the associated radiation doses (discussed in Section 3.3.14)based on these measured values are as follows:

Radionuclide	**Estimated annual release,* Ci**	**Estimated annual dose at nearest residence, mrem**
Gases		
12.3 -yr ^{3}H (as HT)	4.6×10^{-2}	3.1×10^{-7}
(as HTO)	1.6×10^{2}	3.5×10^{-3}
5730 -yr ^{14}C (total)	5.6×10^{-1}	2.5×10^{-4}
4.48-h ^{85m}Kr	6.6	3.0×10^{-3}
10.7 -yr ^{85}Kr	1.7×10^{3}	2.6×10^{-2}
1.27-h ^{87}Kr	1.0×10^{1}	2.2×10^{-2}
2.80-h ^{88}Kr	1.7×10^{1}	2.7×10^{-2}
2.25-d ^{133m}Xe	1.9×10^{1}	3.0×10^{-3}
5.29-d ^{133}Xe	2.0×10^{3}	3.1×10^{-1}
9.15-h ^{135}Xe	5.8×10^{1}	2.5×10^{-2}
Particles and ^{131}I		
313 -d ^{54}Mn	1.1×10^{-4}	1.5×10^{-5}
2.7 -yr ^{55}Fe	1.5×10^{-3}	6.5×10^{-6}
71.3 -d ^{58}Co	6.7×10^{-4}	4.4×10^{-5}
5.26-yr ^{60}Co	1.9×10^{-4}	8.5×10^{-5}
50.5 -d ^{89}Sr	1.3×10^{-5}	3.6×10^{-6}
28.5 -yr ^{90}Sr	9.7×10^{-6}	2.6×10^{-6}
8.06-d ^{131}I	1.4×10^{-2}	4.1×10^{-4}
2.07-yr ^{134}Cs	1.3×10^{-4}	4.5×10^{-5}
30.0 -yr ^{137}Cs	1.9×10^{-4}	5.1×10^{-5}

* For ^{3}H, ^{14}C, kryptons and xenons, the annual release represents the sum of measured individual pathways; the annual release of particles and ^{131}I was computed from the sum of measured primary vent stack discharges during continuous and stored gas releases.

Because these release values are based on occasional—sometimes single—measurements, they can only approximate the total discharges. Whether they are at all applicable was checked by comparing (1) measurements of the same pathway at several points, as in Section 3.3.10; (2) discharge data reported by the station for the two semi-annual periods from July 1, 1970, to June 30, 1971 (see Appendix B.3); and (3) discharge estimates in the Environmental Statement.(8) The latter two are as follows:

Radionuclide	**Annual discharge, Ci**	
	Haddam Neck reports	**Environmental Statement estimate**
^{3}H	0.88	--
^{14}C	--	--
^{85m}Kr	--	23
^{85}Kr	21	510
^{87}Kr	--	4
^{88}Kr	--	24
^{133m}Xe	--	20
^{133}Xe	3,308	2,200
^{135}Xe	225	30
^{131}I	0.012	0.36

The station also reported values for ^{135m}Xe, and the Environmental Statement also contains estimates for ^{83m}Kr, ^{89}Kr, ^{131m}Xe, ^{135m}Xe, ^{137}Xe, ^{138}Xe, and ^{133}I.

The annual releases computed from the measurements in this study are similar to the ^{131}I and ^{133}Xe values reported by the station, but ^{3}H and ^{85}Kr releases are higher and ^{135}Xe releases are lower. The ^{3}H and ^{85}Kr totals from this study are associated mostly with ventilating air discharged from the turbine and vapor containment buildings, respectively. Additional measurements to resolve the differences between measured and reported values would be desirable.

The release values from this study for the radioactive noble gases are all within a factor of four of the Environmental Statement estimates, and are almost identical for the major constituent, ^{133}Xe. Such agreement may be fortuitous, however: on the one hand, the leakage rate from fuel into the reactor coolant assumed in the Environmental Statement estimate was 0.25 percent compared to the actual rate of 0.02 percent;(8) on the other hand, the leakage rates of the radioactive noble gases from the reactor coolant calculated from the measurements reported in Section 3.3.5 are higher than the assumed water leakages. The much higher estimate of ^{131}I releases is mainly due to the noncondensable component of the blowdown flash tank, as described in Section 3.3.12. Direct measurements of ^{131}I in this effluent and of the radioactive noble gases leaking from the reactor coolant would be useful in checking the computational model.

Annual ^{14}C discharges at PWR stations of 11 and 6 Ci per 1,000 MWe have been estimated on the basis of formation rates (19) and measurements,(20) respectively. When reduced 2-fold for the smaller size of the Haddam Neck station, these are still larger than the value of 0.56 Ci given above. The discharge measured in liquid waste (see Section 4.3.3) adds only 0.03 Ci to this value. Some additional gaseous ^{14}C may have been discharged, however, in ventilating air from the primary auxiliary building and the turbine building, for which measured values were <2 and <10 Ci/yr, respectively.

In comparable measurements of airborne effluents at the Yankee-Rowe reactor,(14) discharges of ^{3}H and ^{14}C were generally consistent with the 3-fold smaller size of that reactor. Discharges of the radioactive noble gases were higher by factors of 10^{3} to 10^{4} than at Yankee-Rowe. These higher values suggest higher leakage rates from the fuel into the reactor coolant at the Haddam Neck station.

3.3.14 *Estimated population radiation dose.* The annual whole-body dose to an adult at the residence nearest the Haddam Neck station was 0.5 millirem (mrem) from airborne effluents, according to the values listed in Section 3.3.13. Of the total, 0.3 mrem was from ^{133}Xe. Less than 0.001 mrem was to specific organs from ^{131}I and airborne radioactive particles. The annual dose from inhaling ^{131}I at the maximum ratio of dose to intake—for the 4-year-old—would be four times the listed value, i.e., 0.0016 mrem.(11) Added to the dose values based on measurements of the indicated radionuclides at the station should be small increments due to other radionuclides that are expected to be present, although they were not measured. These, discussed in Sections 3.3.1 to 3.3.12, include ^{41}Ar, ^{131m}Xe, relatively short-lived radioactive noble gas fission products, particulate progeny of the noble gases, and iodine isotopes.

The annual dose was obtained for each listed radionuclide by computing the centerline concentration in ground-level air at the point of interest and then converting from concentration in air to tissue dose. To compute the centerline concentration in ground-level air, the estimated annual discharge was divided by 3.15 x 10^{7} s/yr to obtain the average discharge rate. This rate was multiplied by the relative concentration χ/Q (see Appendix D.1). The relative concentration was computed from the value of $\chi\bar{u}/Q$ plotted as a function of distance in Appendix D.2, the annual average wind velocity, $\bar{u}$, and the directional frequency given in Appendix D.1. The curve for $\chi\bar{u}/Q$ was derived by the station operator from the Gaussian plume model; the meteorological values for the model and the calculations were obtained by the operator during a 15-month measurement period before the station was built. The conversion factors from annual average radionuclide concentration in ground-level air to annual dose are given in Appendix E.1.

The dose to persons would be lower than the computed value at a given location because no adjustment was made for the distribution of the effluent across wind-rose sections, for shielding, and for occupancy factors. Other potential sources of error are associated with the effluent, which is neither at constant concentration nor discharged from a single location; the variable meteorological conditions; and the uneven terrain. These may be reflected (see Appendix D.1) in the 2-to 11-fold lesser dispersion factors used in the Environmental Statement and the greater dispersion factors obtained by the operator in tracer tests and given in the Environmental Report. The EPA dose estimate for a model PWR station with no waste-gas storage holdup in a river valley, reduced by a factor of four because of the lower power level of the Haddam Station, is 0.9 mrem/yr at the nearest residence,(11) within a factor of two of the reported dose.

The radiation doses at other nearby locations listed in Appendix D.1 are all lower than the values given in Section 3.3.13. The annual doses relative to the dose at the nearest residence are:

Location	Distance & direction	Annual dose relative to unity dose at nearest residence
nearest residence	0.7 km WNW	1.00
nearby residence	0.8 km NW	0.82
nearby population group	1.2 km E	0.57
" " "	1.2 km SSE	0.32
fishing in canal	0.5 km SE	0.15

Included above is the relative dose to persons fishing for 500 hours (0.057 yr) on site at the coolant water effluent canal; doses for similarly brief occupancy factors can be computed from the values in Appendix D.1 for locations at the station boundary and on or by the Connecticut River.

3.4 References

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4. RADIONUCLIDES IN LIQUID WASTES

4.1 Liquid Waste System

4.1.1 *Waste processing.*(1) Four categories of radioactive liquid waste are processed at the Haddam Neck station: hydrogen-bearing ("hydrogenated") and air-bearing ("aerated") liquid from the reactor system, and steam generator blowdown and leakage water from the secondary system. The sources of these wastes are listed in Figure 4.1; the processing systems at the time of the study (an augmented system is being installed)(2) are shown in Figure 4.2; and the amounts of liquid handled per year are estimated in Section 2.1.5.

The hydrogenated liquid waste is processed by the boron recovery system. It is mostly "letdown" from the chemical and volume control system, in amounts that fluctuate with reactor operation. Because the "letdown" first flows through the reactor coolant purification demineralizer, radionuclides are removed from the reactor coolant water by demineralizing and filtering. Other portions of reactor coolant water, however, pass untreated into the boron recovery system from reactor coolant drains that collect equipment leaks. The hydrogenated water flows into the primary drain collecting tank and from there is pumped to the waste storage tanks for batch processing. Radioactive decay for 30 days during accumulation has been assumed.(2) The liquid is then passed through a filter (5-micron pore size) and a boric-acid-conditioned cation demineralizer for purification, and evaporated for boron recovery.

The two-stage boron recovery evaporator is designed to operate normally as two units in series at the rate of 75 liters/min (20 gpm) or, occasionally when needed, in parallel at 150 liters/min. Each stage contains a cyclone separator and an entrainment eliminator. The system size is based on its ability to store and process liquid wastes produced in the following plant operations: (a) a refueling startup for each core cycle; (b) a cold shutdown and restart immediately following initial full power operation of each core cycle; (c) one 50 percent, 60-hour load reduction per week; (d) one hot shutdown of at least 60 hours duration every four weeks; (e) one cold shutdown at the end of full power life and one at refueling; and (f) core stretch-out of three months duration at zero boron concentration. The system must also be capable of handling the dilution from two sequential hot shutdowns, with at least 60 hours at maximum power between each shutdown, occurring at any time over the core life. The volume of liquid waste to be processed under extreme conditions includes 45,000 liters from dilution for cold shutdown to full power at the beginning of core life and 55,000 liters from dilution for cold shutdown to full power just before the end of core life.

The steam from the evaporator is condensed and collected in the distillate accumulator. From this small tank, the liquid is pumped to the two test tanks. Conditions in the evaporator are maintained so that most of the boron remains in the evaporator bottoms. These are usually recovered to reuse the boric acid; infrequently, the bottoms are packaged as solid waste for disposal off-site.

The aerated liquid waste contains reactor coolant water that had leaked from the system and collected in various sumps and drains, considerably diluted by process water. Waste water from the reactor and fuel cavity during refueling is also processed as aerated waste. The liquid is collected in two small aerated drain tanks, and then purified by pumping through a prefilter (25-micron pore size) and mixed-bed demineralizer into the same test tanks used for evaporator distillate.

Waste liquids from the reactor system are held in the test tanks for periodic discharge. Radioactive decay for four days during accumulation has been assumed.(2) The liquid is sampled and analyzed for radionuclide content before discharge, and can be recycled for additional treatment, if necessary. For discharge, it is pumped at a selected flow-rate by two 150-liter/min (40 gpm) pumps into the service water discharge header, and then into the circulating coolant water discharge canal.

Blowdown water from the four steam generators in the secondary system is monitored for radioactivity as it flows into the blowdown flash tank, where approximately 35 percent of the water is vented to the

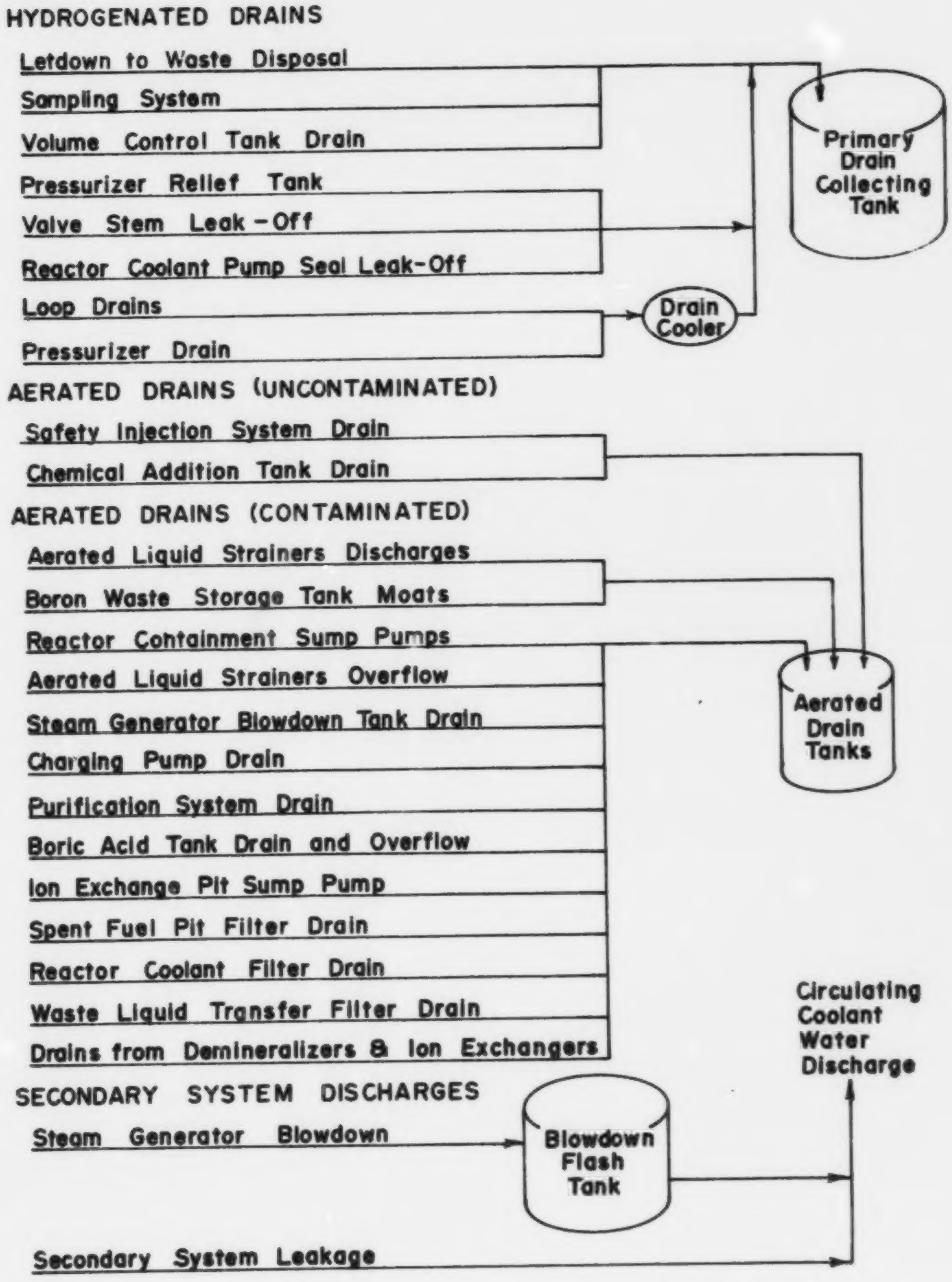
HYDROGENATED DRAINS
Letdown to Waste Disposal
Sampling System
Volume Control Tank Drain
Pressurizer Relief Tank
Valve Stem Leak-Off
Reactor Coolant Pump Seal Leak-Off
Loop Drains
Pressurizer Drain
Drain Cooler
Primary Drain Collecting Tank
AERATED DRAINS (UNCONTAMINATED)
Safety Injection System Drain
Chemical Addition Tank Drain
AERATED DRAINS (CONTAMINATED)
Aerated Liquid Strainers Discharges
Boron Waste Storage Tank Moats
Reactor Containment Sump Pumps
Aerated Liquid Strainers Overflow
Steam Generator Blowdown Tank Drain
Charging Pump Drain
Purification System Drain
Boric Acid Tank Drain and Overflow
Ion Exchange Pit Sump Pump
Spent Fuel Pit Filter Drain
Reactor Coolant Filter Drain
Waste Liquid Transfer Filter Drain
Drains from Demineralizers & Ion Exchangers
Aerated Drain Tanks
SECONDARY SYSTEM DISCHARGES
Steam Generator Blowdown
Blowdown Flash Tank
Secondary System Leakage
Circulating Coolant Water Discharge

Figure 4.1 Sources of Liquid Waste

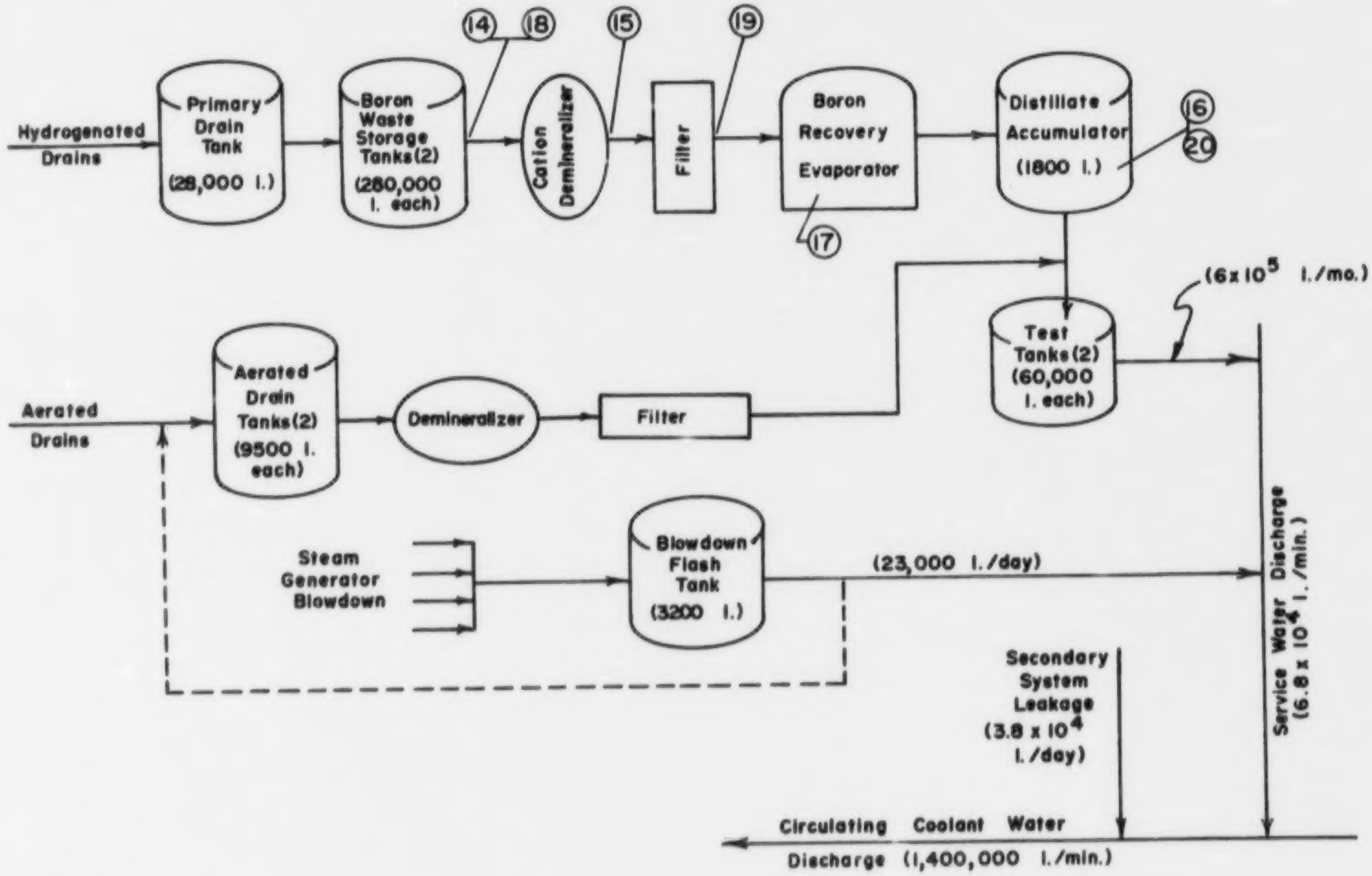
Hydrogenated Drains
Primary Drain Tank (28,000 l.)
Boron Waste Storage Tanks(2) (280,000 l. each)
14
18
Cation Demineralizer
15
Filter
19
Boron Recovery Evaporator
17
Distillate Accumulator (1800 l.)
16
20
Aerated Drains
Aerated Drain Tanks(2) (9500 l. each)
Demineralizer
Filter
Test Tanks(2) (60,000 l. each)
(6x10⁵ l./mo.)
Steam Generator Blowdown
Blowdown Flash Tank (3200 l.)
(23,000 l./day)
Secondary System Leakage (3.8 x 10⁴ l./day)
Service Water Discharge (6.8 x 10⁴ l./min.)
Circulating Coolant Water Discharge (1,400,000 l./min.)

Figure 4.2 Boron Recovery and Liquid Waste Disposal System.
(Numbers Refer to Sampling Locations)

atmosphere as steam.(2) The flow consists of a relatively large nightly blowdown plus a small continuous blowdown (see Section 2.1.5). The liquid waste containing the nonvolatile radionuclides is pumped into the service water discharge header, and from there into the circulating coolant water discharge canal.

The relatively large volume of water leaking from the secondary system (see Section 2.1.5) is collected in turbine building drains. The water is assumed to be mostly from leaking steam and condensate leakage, primarily around pump seals, and to contain all radionuclides except ^{3}H at much lower concentrations than blowdown water. The water flows directly to the circulating coolant water discharge canal.

4.1.2 *Radionuclide release.* The following radionuclides were discharged at Haddam Neck between 1970 and 1973:(3,4)

Radionuclide	Amount in liquid waste, Ci/yr 1970	1971	1972	1973	Limit, Ci/yr**
^{3}H	7380	5830	5890	3900	2×10^{6}
^{54}Mn	0.097†	0.40	0.02	0.02	7×10^{4}
^{58}Co	3.94	0.80	0.97	0.76	7×10^{4}
^{60}Co	0.013	0.81	1.15	0.59	2×10^{4}
^{131}I	0.76	2.1	0.30	0.05	2×10^{2}
^{133}I	0.13†	1.0	0.57	0.15	7×10^{2}
^{133}Xe	15.4	29.9	7.35	1.16	–
^{135}Xe	0.041	0.16	0.17	0.05	–
^{137}Cs	0.10	0.62	0.71	0.31	2×10^{4}
Gross beta-gamma*	6.7	5.7	4.8	3.0	7×10^{1}
Volume (liters)	23.5×10^{6}	28.7×10^{6}	34.4×10^{6}	26.8×10^{6}	–

* Does not include ^{3}H, ^{133}Xe, and ^{135}Xe; in addition, 0.025 Ci ^{59}Fe was discharged in January – June, 1971.

**Discharge into circulating cooling water flowing at the rate of 7.3×10^{14} ml/yr.

† Does not include values for the first 6 months.

The releases for 1970 and 1971 are typical of the station, except that values were very low during the first year of operation in 1967. For the period 1967 through 1973, annual discharges of ^{3}H ranged from 120 to 7380 Ci, and of gross beta activity, from 0.4 to 12 Ci.(3,4)

The average concentration of radionuclides in the circulating coolant discharge canal due to station releases can be calculated at dilution volumes for the years 1970 and 1971 of 6.5×10^{11} liters and 7.3×10^{11} liters, respectively, as follows:(3,4)

1970: $\mu Ci/ml$ = amount released $(Ci/yr) \times 1.5 \times 10^{-9}$

1971: $\mu Ci/ml$ = amount released $(Ci/yr) \times 1.4 \times 10^{-9}$

During these two years, the average gross beta radioactivity at the point of discharge was, therefore, approximately 1×10^{-8} µCi/ml, and the ^{3}H concentration, 1×10^{-5} µCi/ml. The average concentrations apply after complete mixing in the canal; further dilution occurs beyond the mouth of the coolant discharge canal in the Connecticut River. Instantaneous concentrations would have fluctuated considerably, depending on the liquid waste being discharged.

Concentrations of radionuclides in effluents to unrestricted areas are limited by the AEC according to paragraph 20.106 of 10 CFR 20. Concentrations above background in water, averaged over 1 year, as listed in Appendix B, Table II, column 2 of 10 CFR 20, are applied at the boundary of the restricted area. The limit is 1×10^{-7} µCi/ml for an unidentified mixture containing no ^{129}I, ^{226}Ra, and ^{228}Ra. Limits for individual radionuclides are 3×10^{-3} µCi/ml for ^{3}H, the radionuclide at highest concentration in Haddam Neck effluent, and 3×10^{-7} µCi/ml each for soluble ^{90}Sr and ^{131}I, usually the radionuclides with the lowest limits in reactor effluent. Higher limits are permissible under conditions of Subsection (b) of paragraph 20.106, or more stringent limits may be applied under Subsection (e).

The limits at the coolant canal discharge for an annual flow rate of 7.3×10^{14} ml are tabulated above with the radionuclide discharges. The gross beta values were less than 10 percent of the limits and the individual radionuclides were at or below 1 percent.

4.2 Samples and Analyses

4.2.1 *Samples.* The following samples* of water from the station were provided by station staff:

11. test tank during discharge, 1 liter, collected Sept. 15, 1970 at 0900;
12. test tank during discharge, 3.5 liters, collected Mar. 15, 1971 at 1000;
13. aerated drain tank, 3 liters, collected July 23, 1970 at 1510;
14. evaporator feed from boron waste storage tank, 1 liter, collected Feb. 5, 1971 at 1030;
15. evaporator feed after demineralizer, 1 liter, collected Feb. 5, 1971 at 1035;
16. evaporator distillate before test tank, 1 liter, collected Feb. 5, 1941 at 1045;
17. boron product from evaporator (boric acid mixture), 1 liter, collected Feb. 5, 1971 at 1100; these evaporator bottoms were from previous batches;
18. evaporator feed from boron waste storage tank, 1 liter, collected May 25, 1971 at 1125;
19. evaporator feed after demineralizer and filter, 1 liter, collected May 25, 1971 at 1130;
20. evaporator distillate after first stage, 4 liters, collected May 25, 1971 at 1145;
21. aerated drain tank water after demineralizer, 4 liters, collected May 25, 1971 at 1135.

The efficacy of waste treatment in the boron recovery system was observed with two sets of samples. On February 5, 1971, samples of liquid waste were collected before treatment (#14), after passage through the cation-exchange resin (#15), and from the condensate after evaporation (#16). The concentrated borate solution—the evaporator bottoms—could not be collected, but an earlier boron batch (#17) was sampled. On May 25, 1971, samples of liquid waste were collected before treatment (#18), after ion exchange and filtration (#19), and from the condensate after evaporation in the first stage (#20).

Aliquots of liquid from the aerated drain tanks were obtained on July 23, 1970, and May 25, 1971 (samples #13 and 21). Samples of liquid waste just before discharge were obtained on Sept. 15, 1970, and March 15, 1971 (samples #11 and 12). The wastes were discharged from the test tank into the circulating coolant water discharge canal at flow rates of 19 liters/min (5 gpm) and 76 liters/min (20 gpm), respectively. Six samples of steam generator blowdown water—samples #4 to 9 listed in Section 2.1.7— were also analyzed. No samples of water leaking from the secondary system were obtained.

To confirm discharge values computed from radionuclide measurements in liquid wastes, the following large samples of river water were collected and analyzed for radionuclide content:

Location	Time	Volume, liters	Suspended solids, mg/l	Hardness, mg $CaCO_3$/l
Discharge on Sept. 15, 1970, 0905 – Sept. 16, 1130				
Service water intake	0930	164	60	50
Discharge weir	1030	204	130	—
Canal mouth	1115	204	140	—
Discharge on March 15, 1971, 1020 – 2400				
Service water intake	0915	148	6.3	40
Discharge weir	1000	194	7.3	—
Discharge weir	1045	201	5.4	—
Canal mouth	1150	194	6.1	—
Service water intake	1245	144	4.0	—

Service water is taken from influent circulating coolant water; the canal mouth is 1.7 km downstream from the weir in the circulating coolant water canal. The first and second samples on March 15 were obtained to measure radionuclides in blowdown and leakage from secondary system; the others, to measure the combined radionuclides discharged from the secondary system and reactor system test tanks.

4.2.2 *Analysis of waste solutions.* The liquids from the station were analyzed spectrometrically with a Ge(Li) gamma-ray detector. The samples were first counted within a week after collection and again several months afterwards to identify radionuclides by combining observations of gamma-ray energies and decay rates. The identified radionuclides were quantified by computing disintegration rates from count rates under characteristic photon peaks on the basis of prior counting efficiency calibrations of these detectors. The samples were analyzed radiochemically for ^{3}H, ^{14}C, ^{32}P, ^{55}Fe, ^{63}Ni, ^{89}Sr, and ^{90}Sr.(5) If short-lived radionuclides such as ^{24}Na and ^{133}I were present, they were not detected because of the relatively long period between collection and analysis.

4.2.3 *Analysis of circulating coolant water.* The water volumes shown in Section 4.2.1 (148–204 liters) were collected in 210-liter drums and all but 4 liters were passed through 5-section ion-exchange columns(6) at flow rates of approximately 100 ml/min. Each of the columns was then separated into 3 cation-exchange resin sections, 2 anion-exchange resin sections, and a glass-wool filter.(6) Each part was analyzed with a NaI(Tl) gamma-ray spectrometer for 1,000-minute counting periods. The anion-exchange resins were recounted at weekly intervals to confirm the ^{131}I measurements. Every cation-exchange resin section was eluted with 1,200 ml 6N HCl. The elutriants were analyzed radiochemically in sequence for strontium, cesium, and cobalt. The columns appeared to be sufficiently large to retain the ionic radionuclides, in that only 5 to 10 percent of the radionuclide amounts measured in the top cation- or anion-exchange resin section were found on the second section. No radionuclides could be detected on the third cation-exchange resin section.

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*Samples No. 1 – 10 are described in Section 2.1.7.

The remaining 4 liters of water were analyzed for hardness (calcium plus magnesium) and some radionuclides. Ten-ml aliquots were used to determine tritium (see Section 5.2.1) and hardness. Most of the 4-liter sample was acidified with 10 ml conc. HNO_3 and evaporated to 45 ml. One-third of this sample was evaporated to dryness and analyzed with a NaI(Tl) gamma-ray spectrometer. The other 30 ml were analyzed sequentially for radioactive strontium, cesium, and cobalt, but radionuclide concentrations were usually too low for comparing results with those obtained with the cation-exchange resins. These radionuclides were counted for 100– or 1000-min periods with G-M detectors at a background of approximately 1.5 counts/min.

Solids that had settled in each drum were collected by slurrying with the 4 liters of water, and combined with solids flushed from the glass wool filter. They were filtered, dried at 110° C, and weighed to determine the amount of suspended solids. These samples were then analyzed by a NaI(Tl) gamma-ray spectrometer.

4.3 Results and Discussion

4.3.1 *Radionuclides in the boron recovery system.* The fraction of radionuclides removed by successively filtering, demineralizing, and evaporating two batches of hydrogenated liquid waste was smaller than is generally reported for such processes. The overall decontamination factor (DF, defined as the concentration in the influent stream divided by the concentration in the effluent) for the liquid on Feb. 5, 1971, based on the concentration measurements shown in Table 4.1, was less than 100 for all radionuclides except radiocesium. Overall DF values were much higher for the batch of May 25, 1971, but only the DF values for radio-manganese and -cobalt exceeded 10,000.

In contrast, the AEC has computed removals of nonvolatile radionuclides from waste streams with the following DF values:(7,8)

	Decontamination factor	
Radioelement	**cation-exchange resin**	**boric acid evaporator**
iodine	**1**	**100**
other anions	**1**	**1,000**
cesium, rubidium	**10**	**1,000**
other cations	**100**	**1,000**

Surveys of waste treatment practices in the nuclear industry also report claims of much higher decontamination factors by these processes.(8-11) (Note, however, that decontamination factors for evaporators are sometimes reported as the ratio of the concentrate to the distillate, which may be much higher than the DF relative to the feed).

The DF values computed separately in Table 4.2 for the demineralizer and the evaporator are, with a few exceptions, distinctly different for the two batches of waste. The DF value of unity (no decontamination) for ^{3}H and ^{14}C is to be expected if these radionuclides are in the form of water and carbon dioxide, respectively. The DF values for radio-iron, -cobalt, and -iodine at the evaporator were also consistently low. One reason for the differences in DF values was probably an unpremeditated change in sampling-point selection that resulted in including filtration with evaporation in one batch, and with demineralization, in the other. The two liquid wastes were also different in the content, and possibly the chemical form, of radionuclides. The liquid processed on Feb. 5 contained ^{3}H at concentrations similar to reactor coolant water. Most other radionuclides were at lower concentrations (see Tables 4.1 and 2.1), presumably due to the removal of ionic radionuclides by the reactor coolant purification demineralizer and radioactive decay during waste accumulation. The feed solution on May 25 had a lower ^{3}H concentration and higher values for all other radionuclides. It was probably from reactor shut-down for refueling. Process operating parameters also may have been changed between the two batches.

The concentrations of radionuclides in the boron product of Feb. 5 (see Table 4.1) refer to a batch treated earlier, and are, therefore, only qualitative indications of the extent to which radionuclides are recirculated to the reactor coolant with recovered boron. In this instance, ^{3}H, followed by ^{55}Fe, ^{58}Co, and ^{60}Co, were the radioactive constituents at highest concentrations.

4.3.2 *Radionuclides in aerated liquid waste.* The sample of July 23, 1970, contained, in very different proportions, the radionuclides of the July 24 sample of reactor coolant water. Concentrations of ^{3}H, ^{90}Sr, and ^{131}I, *for example, were lower by approximately two* orders of magnitude, while ^{51}Cr, ^{60}Co, ^{106}Ru, and ^{110m}Ag were at considerably higher concentrations (see Table 4.3). Reduction of some of these radionuclide

Table 4.1

Radionuclide Concentration in Boron Recovery System, pCi/ml

	February 5, 1971				May 25, 1971		
Nuclide	Evaporator Feed From Storage Tank (14)*	Evaporator Feed After Demineralizer (15)	Evaporator Distillate Before Test Tank (16)	Boron Product (17)	Evaporator Feed From Storage Tank (18)	Evaporator Feed After Demineralizer and Filter (19)	Evaporator Distillate First Stage (20)
^{3}H	2.8 x 10^{6}	2.8 x 10^{6}	2.8 x 10^{6}	3.6 x 10^{5}	1.6 x 10^{5}	1.6 x 10^{5}	1.5 x 10^{5}
^{14}C	13	18	17	3.1	3.7	0.9	1.2
^{32}P	<0.1	---	---	0.2	18	NA	<0.1
^{51}Cr	<1	---	---	---	180	<0.5	<0.5
^{54}Mn	2.9	0.5	0.14	150	1400	~0.6	<0.1
^{55}Fe	7.4	1.9	1.1	4800	2000	2.3	0.7
^{59}Fe	0.2	<0.2	---	20	170	0.4	0.14
^{57}Co	0.21	<0.03	---	10	44	<0.04	<0.03
^{58}Co	15	0.7	0.18	620	6300	4.4	0.3
^{60}Co	6.4	1.0	0.5	520	2700	1.0	0.4
^{89}Sr	<0.05	---	---	<0.05	75	<0.03	<0.03
^{90}Sr	<0.05	---	---	0.19	2.8	<0.03	<0.03
^{95}Zr	<1	---	---	~3	120	<0.4	0.4
^{95}Nb	<1	---	---	~3	140	<0.1	<0.1
^{103}Ru	<0.1	---	---	~2	28	0.2	<0.2
^{106}Ru	<1	---	---	---	12	<1	<1
^{110m}Ag	0.47	<0.1	---	11	3.6	<0.1	<0.1
^{124}Sb	<0.1	---	---	13	57	<0.1	<0.1
^{131}I	0.45	0.30	0.14	0.12	1400	51	48
^{134}Cs	3.6	1.3	<0.03	16	13,000	27	15
^{137}Cs	3.6	1.3	<0.03	21	11,000	24	14
^{140}Ba	0.2	<0.1	---	<0.1	100	<0.1	<0.1
^{141}Ce	<0.1	---	---	---	39	<0.2	<0.2
^{144}Ce	<0.2	---	---	---	37	<0.4	<0.4

* Numbers in parentheses refer to sample numbers in Section 2.1.7 and in Figure 4.2.

Notes:

1. ^{133}Xe was detected in all feed and distillate samples.
2. 1 pCi/ml = 10^{6} µCi/ml
3. NA – not analyzed

Table 4.2
Decontamination Factors for Demineralizing, Demineralizing plus Filtering, and Evaporating

	February 15, 1971		May 25, 1971	
Nuclide	Demineralizer	Filter plus Evaporator	Demineralizer plus filter	Evaporator, First stage
^{3}H	1	1	1	1
^{14}C	1	1	4	1
^{51}Cr	---	---	> 400	--
^{54}Mn	6	4	2300	>6
^{55}Fe	4	2	900	3
^{59}Fe	>1	---	400	3
^{57}Co	>7	---	>1100	--
^{58}Co	21	4	1400	15
^{60}Co	6	2	2700	3
^{89}Sr	---	---	>2500	--
^{90}Sr	---	---	> 90	--
^{95}Zr	---	---	> 300	--
^{95}Nb	---	---	>1400	--
^{103}Ru	---	---	140	>1
^{106}Ru	---	---	> 12	--
^{110m}Ag	>5	---	> 360	--
^{124}Sb	---	---	> 600	--
^{131}I	2	2	30	1
^{134}Cs	2	>40	500	2
^{137}Cs	3	>40	500	2
^{140}Ba	2	---	>1000	--
^{141}Ce	---	---	> 190	--
^{144}Ce	---	---	> 90	--

Note:

Values computed from concentrations in Table 4.1.

A value of 1 indicates no decontamination.

concentrations before the liquid is discharged would be expected from the demineralization treatment that follows collection in the drain tank. The demineralized sample of May 25, 1971, was from liquid waste collected during refueling, and resembles the evaporator feed in radionuclide content after demineralization (see column 7 of Table 4.1).

4.3.3 *Radionuclide discharge to circulating coolant water.* In the two samples of reactor system liquid waste collected just before the test tanks were discharged (see Table 4.4), radionuclide concentrations were somewhat lower, on the average, than in reactor coolant water (see Table 2.1). Although the concentration of radionuclides in reactor coolant water and liquid waste are not directly comparable because the samples were collected at different times, ^{3}H concentrations suggest a several-fold dilution of reactor coolant water; ^{133}I and ^{131}I concentrations indicate radioactive decay for about 1 week; and concentrations of radio-carbon, -manganese, -iron, -cobalt, -strontium, and -iodine suggest relatively low DF values in processing the reactor system liquid waste, as observed in Section 4.3.1.

The amount of radionuclides discharged annually from the reactor system in liquid waste was estimated in Table 4.4 to be 8,000 Ci ^{3}H, 20 Ci ^{133}Xe, 4 Ci ^{131}I, and 1.3 Ci of all others. The totals were computed by multiplying the averages of the two sets of measured radionuclide concentrations in test tank liquid by the waste liquid discharge of 5 x 10^{6} liters reported for 1971 (Appendix B.3). The discharge would be approximately 10 percent less in ^{3}H and ^{131}I if lower concentrations during refueling were taken into account (see samples of May 25, 1971 in Tables 4.1 and 4.3). Discharges of ^{134}Cs and ^{137}Cs, on the other hand, would be somewhat higher.

Although these estimated values are based on only two sets of measurements, they are consistent with discharge data reported by the station operator for July – December 1970 (Appendix B.4) except that the ^{131}I

Table 4.3

Radionuclide Concentration in Aerated Liquid Waste, µCi/ml

Radionuclide	Drain Tank July 23, 1970	After Demineralizing May 25, 1971
^{3}H	3.2×10^{-1}	1.4×10^{-1}
^{14}C	5.5×10^{-6}	1.0×10^{-6}
^{32}P	3.1×10^{-6}	$<5 \times 10^{-7}$
^{51}Cr	1.3×10^{-3}	$<5 \times 10^{-7}$
^{54}Mn	3.5×10^{-3}	1.6×10^{-6}
^{55}Fe	2.8×10^{-2}	8.0×10^{-7}
^{59}Fe	9.6×10^{-5}	2.9×10^{-7}
^{58}Co	3.0×10^{-2}	1.8×10^{-5}
^{60}Co	9.0×10^{-3}	3.5×10^{-6}
^{89}Sr	1.0×10^{-7}	4.0×10^{-8}
^{90}Sr	2.0×10^{-7}	3.0×10^{-8}
^{95}Zr	2.0×10^{-7}	2.6×10^{-7}
^{95}Nb	1.5×10^{-5}	5.0×10^{-8}
^{106}Ru	1.0×10^{-5}	$<4 \times 10^{-6}$
^{110m}Ag	1.7×10^{-5}	3×10^{-7}
^{124}Sb	5.8×10^{-6}	1.5×10^{-7}
^{131}I	1.0×10^{-6}	3.7×10^{-6}
^{134}Cs	3.5×10^{-5}	4.8×10^{-5}
^{137}Cs	5.0×10^{-5}	4.4×10^{-5}
^{140}Ba	1.3×10^{-6}	$<1 \times 10^{-7}$
Gross alpha	1.8×10^{-8}	NA

Notes:

1. Not detected ($<1 \times 10^{-6}$ pCi/ml): ^{57}Co, ^{103}Ru, ^{136}Cs, ^{141}Ce, and ^{144}Ce.
2. NA: not analyzed

value is approximately twice as high as the operator's. The operator also reported discharging small amounts of ^{133}Xe. In the Environmental Statement, the estimated liquid "radwaste" discharge has been normalized to a total (not including ^{3}H and gases) of 5 Ci/yr, but contains radionuclides in very different proportions:(2)

^{99}Mo	0.38 Ci/yr	^{136}Cs	0.47 Ci/yr
^{99m}Tc	0.37	^{137}Cs	1.2
^{131}I	0.22	^{137m}Ba	1.2
^{134}Cs	1.2	other	0.16

A total ^{3}H discharge of 8,000 Ci/yr was estimated in the Environmental Statement, as in this report.

The averages of concentrations measured in six samples of steam generator blowdown water and the estimated annual discharges are listed in Table 4.5. Where concentrations were too low to be measured, averages were based on the values computed in Appendix C.3. To compute the discharges, average concentrations of all radionuclides except ^{3}H were multiplied by the estimated blowdown flow of 6×10^{6} kg/yr (Section 2.1.5). For ^{3}H, the estimated amount of steam flashing in the blowdown flash tank was subtracted from the water volume, but the amount of water leaking from the secondary system was taken into account in the last column of Table 4.5.

The amount of radionuclides discharged annually in liquid waste from the secondary system was estimated to be 260 Ci ^{3}H, 2 Ci each of ^{131}I and ^{133}I, 0.8 Ci ^{135}I, and 0.3 Ci of all other radionuclides with half-lives of 6 hours or longer. The station operator reported (Appendix B.4) similar discharges of ^{3}H and gross beta activity in July – December 1970, but less ^{131}I and ^{133}I. In the Environmental Statement,(2) non-tritium discharges were estimated to be approximately twice the total in Table 4.5:

^{58}Co	0.44 Ci/yr	^{133}I	3.5 Ci/yr
^{88}Rb	0.10	^{135}I	1.1
^{99}Mo	0.11	^{134}Cs	0.24
^{99m}Tc	0.10	^{136}Cs	0.14
^{132}Te	0.33	^{137}Cs	0.29
^{131}I	3.5	^{137m}Ba	0.27
^{132}I	0.52	others	0.3

Table 4.4

Radionuclide Discharge from Reactor System Liquid Waste

	Concentration in test tank, µCi/ml		
Radionuclide	September 15, 1970	March 15, 1971	Estimated discharge, Ci/yr
^{3}H	1.8	1.5	8×10^{3}
^{14}C	2.9×10^{-6}	9.5×10^{-6}	3×10^{-2}
^{32}P	2×10^{-7}	$<2 \times 10^{-7}$	5×10^{-4}
^{51}Cr	1.2×10^{-6}	$<5 \times 10^{-7}$	3×10^{-3}
^{54}Mn	5.0×10^{-5}	2.2×10^{-6}	1×10^{-1}
^{55}Fe	1.3×10^{-4}	1.0×10^{-5}	4×10^{-1}
^{59}Fe	3.4×10^{-6}	$<2 \times 10^{-7}$	8×10^{-3}
^{57}Co	1.3×10^{-6}	1.0×10^{-6}	6×10^{-3}
^{58}Co	1.6×10^{-4}	8.1×10^{-6}	4×10^{-1}
^{60}Co	7.5×10^{-5}	7.6×10^{-6}	2×10^{-1}
^{63}Ni	1.1×10^{-6}	NA	6×10^{-3}
^{90}Sr	1.3×10^{-7}	5×10^{-8}	4×10^{-4}
^{95}Zr	2×10^{-7}	$<1 \times 10^{-7}$	5×10^{-4}
^{95}Nb	$<2 \times 10^{-7}$	1.1×10^{-6}	3×10^{-3}
^{110m}Ag	1.1×10^{-6}	2×10^{-7}	3×10^{-3}
^{131}I	1.8×10^{-3}	1.6×10^{-5}	4
^{133}I	2×10^{-5}	2.2×10^{-5}	1×10^{-1}
^{133}Xe	4.8×10^{-3}	1.8×10^{-3}	2×10^{1}
^{134}Cs	1.0×10^{-6}	3.7×10^{-6}	1×10^{-2}
^{137}Cs	9×10^{-7}	5.1×10^{-6}	2×10^{-2}

Notes:

1. < values are 3-sigma counting error.
2. The following radionuclides were not detected: ($<2 \times 10^{-7}$ µCi/ml, approximately) ^{89}Sr, ^{99}Mo, ^{103}Ru, ^{124}Sb, ^{136}Cs, ^{140}Ba, ^{141}Ce, and ^{144}Ce; ($<1 \times 10^{-9}$ µCi/ml) gross alpha.
3. The annual discharge was estimated by multiplying the average of the two radionuclide concentrations by the annual liquid waste discharge from the reactor system of 5×10^{9} ml (5×10^{6} kg—see Section 2.1.5). The estimated discharge of each radionuclide listed in note 2 would be $<1 \times 10^{-3}$ Ci/yr.
4. NA – not analyzed.

The above estimates are similar to the values of ^{131}I, ^{133}I, and ^{135}I measured in this study; are much higher for ^{58}Co, ^{99}Mo, ^{132}Te, ^{134}Cs, and ^{137}Cs; and provide values for very short-lived ^{88}Rb and ^{134}I, which could not be measured.

The annual discharge of radionuclides to the Connecticut River during the period of study was taken to be the sum of the estimated discharges in Table 4.4 and 4.5. It was assumed that water leaking from the secondary system contained negligible amounts of radionuclides other than ^{3}H, and that all other discharged liquids were nonradioactive. Of the discharged radionuclides, the short-lived ones, ^{32}P and radiocesium were observed mostly in blowdown water; all others were more in reactor-system wastes. The estimated values for the study period during the last half of 1970 and the first half of 1971 are reasonably consistent with the discharges reported by the station operator for 1970 and 1971 (see Section 4.1.2). The amounts of ^{54}Mn, ^{60}Co, ^{133}Xe, and ^{137}Cs are intermediate to the two sets of annual values reported by the operator, those for ^{3}H, ^{131}I, and ^{133}I were higher, and the amount of ^{58}Co was lower. Compared to the Yankee-Rowe station,(12) the same radionuclides generally were discharged in amounts higher by one to two orders of magnitude. Only ^{51}Cr was discharged in higher amounts at Yankee-Rowe.

These amounts of radionuclides in water have no direct health implication because the Connecticut

Table 4.5

Radionuclide Discharge from Secondary System Liquid Waste

Radionuclide	Average concentration, uCi/ml	* Estimated discharge, Ci/yr	
		Blowdown†	Leakage**
^{3}H	1.4×10^{-2}	6×10^{1}	2×10^{2}
^{14}C	1×10^{-7}	6×10^{-4}	
^{24}Na	3×10^{-6}	2×10^{-2}	
^{32}P	1.2×10^{-6}	7×10^{-3}	
^{51}Cr	3×10^{-7}	2×10^{-3}	
^{54}Mn	2.5×10^{-7}	2×10^{-3}	
^{55}Fe	7.4×10^{-6}	4×10^{-2}	
^{59}Fe	(1×10^{-7})††	(6×10^{-4})	
^{57}Co	(3×10^{-8})	(2×10^{-4})	
^{58}Co	3.7×10^{-6}	2×10^{-2}	
^{60}Co	2.0×10^{-7}	1×10^{-3}	
^{63}Ni	(4×10^{-7})	(2×10^{-3})	
^{90}Sr	1.6×10^{-8}	1×10^{-4}	
^{95}Zr	4×10^{-8}	2×10^{-4}	
^{95}Nb	(4×10^{-8})	(2×10^{-4})	
^{99}Mo	4.5×10^{-6}	3×10^{-2}	
^{131}I	3.2×10^{-4}	2	
^{133}I	3.3×10^{-4}	2	
^{135}I	1.4×10^{-4}	8×10^{-1}	
^{134}Cs	1.3×10^{-5}	8×10^{-2}	
^{136}Cs	3.3×10^{-6}	2×10^{-2}	
^{137}Cs	1.4×10^{-5}	8×10^{-2}	

- * Concentrations are averages of measured values from Table 2.2.
- ** Leakage from the secondary system at the estimated annual rate of 1.2×10^{10} ml.
- † The average concentration multiplied by an estimated annual blowdown of 6×10^{6} kg (6×10^{9} ml) for all radionuclides, except 4×10^{6} kg for ^{3}H.
- †† Concentrations in parentheses are averages of computed values from Appendix C.3.

River at and below the station is not a source of public water supply. The intake of radionuclides through eating fish caught in these waters is discussed in Section 5.5.4.

4.3.4 *Radionuclides in circulating coolant water.* The three radionuclides—^{3}H, ^{131}I, and ^{58}Co—computed to be at highest concentration in the liquid effluent were detected on Sept. 15, 1970, in cooling canal water at the weir and mouth, as shown in Table 4.6. The measured concentrations, however, were inconsistent with effluent values. Compared to the predicted totals in Table 4.6, measured concentrations of ^{3}H, ^{58}Co, ^{60}Co, and ^{131}I were approximately 10-fold lower. These low values can be explained if the test tank containing reactor system waste initially was discharged more slowly than reported; this hypothesis is supported by the higher ^{3}H concentrations measured downstream in the Connecticut River later in the day (see Section 5.2.2). Measured concentrations of ^{134}Cs and ^{137}Cs in the canal, and of ^{131}I at the station water intake before flow from the test tank would have reached it, however, require a source other than the test tank. One possibility is that these radionuclides remained from the much more voluminous nightly blowdown; another, that they were in water leaking from the secondary system.

Some of the ^{3}H and ^{137}Cs in the water and most of the ^{90}Sr are attributed to fallout. The following radionuclide concentrations are reported in U. S. surface waters:

Table 4.6

Radionuclide Concentrations in Circulating Coolant Water in September 15, 1970, pCi/liter

Radio-nuclide	Measured			Calculated from discharge		
	Intake	Weir	Canal mouth	Secondary* system	Reactor† system	Total
^{3}H	<0.6x10^{3}	1.2x10^{3}	3.7x10^{3}	0.05x10^{3}	24 x 10^{3}	24 x10^{3}
^{58}Co	<0.1	0.2	0.3	0.02	2.2	2.2
	+0.1**	+0.1	+0.1			
^{60}Co	<0.1	<0.1	<0.1	<0.01	1.0	1.0
^{90}Sr	0.5	0.6	0.7	<0.01	0.002	<0.01
^{131}I	1.0	3.5	3.0	0.5	24	24
		+0.1	+0.1			
^{134}Cs	<0.1	0.3	0.4	0.05	0.01	0.06
^{137}Cs	<0.1	0.4	0.3	0.06	0.01	0.07

* Concentrations in Table 2.2 on the indicated date divided by the following dilution factors:

^{3}H—circulating coolant water flow rate of 1.4 x 10^{6} l/day divided by average discharge (see Section 2.1.5) of 20 l/min (28,700 l/day) from leakage plus 1.0 l/min (0.65 x 2,300 l/day) from continuous blowdown = 1.4 x 10^{6}/21 = 6.7 x 10^{4}

other radionuclides—1.4 x 10^{6} l/day divided by continuous blowdown flow of 1.6 l/min (2,300 l/day) = 8.8 x 10^{5}

**Value after plus indicates radionuclide on suspended solids, calculated per volume of water from amount of suspended solids.

† Concentrations on the indicated date in Table 4.4 divided by dilution factor of 1.4 x 10^{6} l/min ÷ 19 l/min = 7.4 x 10^{4}.

Note: < values are 3-sigma counting error; the 2-sigma counting errors were 0.6 pCi/ml for ^{3}H and 0.1 pCi/l for all other radionuclides.

	Concentration, pCi/liter		
	^{3}H	^{90}Sr	^{137}Cs
Surface water, July 1970 – March 1971(1)	<200–10,300	---	--
Surface water, January – March 1971(10)	---	0.4–2.1	--
Lake Michigan, August – September 1970(10)	400– 900	0.4–2.3	<0.1–0.8

At least the lower extremes of these ranges reflect concentrations of fallout-related radionuclides in surface water.

The detection of ^{131}I in the Connecticut River at the station water intake suggests that some of the cooling water recirculates. River flow near the station reverses during flood tides (see Section 5.1.2), and temperature measurements have shown that warm water discharged into the coolant canal reaches to and beyond the water intake.(2)

To distinguish between radionuclides from blowdown and test tank discharges, a second set of radionuclide measurements in effluent water, undertaken on March 15, 1971, included collection of samples both before and during the test tank discharge. The former samples were expected to contain measurable amounts of ^{3}H and ^{131}I from continuous blowdown; the latter, higher concentrations of these two radionuclides as well as detectable amounts of ^{59}Fe (which was not analyzed for), ^{58}Co, ^{60}Co, ^{134}Cs, and ^{137}Cs. The results shown in Table 4.7 also indicate radionuclides from fallout at the following levels: ^{3}H, ≤ 700 pCi/liter; ^{90}Sr, 0.4–0.6 pCi/liter; and ^{137}Cs, <0.1–0.2 pCi/liter.

When blowdown and leakage from the secondary system were the only known sources at the station of radionuclides in the coolant canal water, ^{58}Co and ^{131}I were measured at the weir at higher concentrations than predicted from discharge values (see Table 4.7). The two radionuclides were also at the station water intake. These observations support the above-cited possibilities that (1) either the nightly blowdown or leakage water resulted in higher concentrations than computed from only the continuous blowdown, or (2) some cooling water recirculates.

Increased levels of ^{3}H, ^{58}Co, ^{131}I, ^{134}Cs, and ^{137}Cs were associated with the discharge of the test tank. Except for ^{58}Co and ^{60}Co, the concentration increases in canal water were consistent with predicted values. A possible reason for the lower radiocobalt concentrations is sorption on suspended material or other surfaces. The presence of some radiocobalt, as well as radioiodine, on suspended material is shown in Table 4.6. The movement of coolant water upstream is indicated by elevated values of ^{3}H at the coolant water intake, as well as the detection there of ^{131}I and ^{134}Cs.

The above observations demonstrate the feasibility of measuring some of the radionuclides in liquid effluents after discharge to observe effects such as the recirculation of water and the removal of radionuclides from water. Improved coordinations with effluent measurements would be needed to compare measured with predicted concentrations. Possible sources of error in the environmental measurements include incomplete mixing of the waste liquid with the circulating coolant water, deposition of radionuclides on surfaces between discharge and sampling locations, formation of chemical forms not retained by the collectors, and uncertainty concerning flow rates.

4.4 References

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Table 4.7

Radionuclide Concentrations in Circulating Coolant Water on March 15, 1971, pCi/liter

	Measured during continuous discharge from secondary system		Measured during continuous and test-tank discharge			Calculated from discharge*	
Radio-nuclide	Intake	Weir	Intake	Weir	Canal mouth	Secondary system	Reactor system
^{3}H	1.0x10^3	1.3x10^3	4.2x10^3	65 x10^3	20 x 10^3	0.3x10^3	81 x10^3
^{58}Co	0.2	0.2	0.2	0.5	0.4	<0.01	0.44
^{60}Co	<0.1	<0.1	<0.1	<0.1	<0.1	0.01	0.41
^{90}Sr	0.5	0.4	0.5	0.5	0.6	0.01	0.003
^{131}I	0.5	2.6	0.6	3.4	3.9	0.6	0.87
		+ 0.1*		+ 0.1			
^{134}Cs	<0.1	<0.1	0.2	0.3	0.2	0.01	0.20
^{137}Cs	0.2	0.2	0.1	0.3	0.3	0.01	0.28

*See footnotes to Table 4.6, except that dilution factor from reactor system was 1.4x10^6 l/min ÷ 76 l/min = 1.8 x 10^4

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5. RADIONUCLIDES IN THE AQUATIC ENVIRONMENT

5.1 Introduction

5.1.1 *Studies near Haddam Neck.* The measurements described in Section 4.3.4 showed that detectable concentrations of radionuclides from the Haddam Neck station were in the water of the discharge canal and the Connecticut River near the station. Efforts to measure effluent radionuclides in the aquatic environment were, therefore, concentrated in these areas. The following studies are described in detail in Sections 5.2 to 5.7:

(1) Tritium was measured in the water of the Connecticut River between Haddam Island State Park and the East Haddam bridge during two releases of radioactive liquid waste by the station. Tritium concentrations above background were found at the mouth of the discharge canal and as far as 2 km downstream and 3 km upstream. Other radionuclides in river water are reported in Section 4.3.4.

(2) Aquatic plants, plankton, and algae from the mouth of the discharge canal and the river near the water intake contained several radionuclides discharged by the station at approximately 10^4 times the computed water concentration. Aquatic plants were not found in the discharge canal, nor algae in the river near the station.

(3) Radionuclides discharged by the station were detected in fish from the discharge canal, but not in fish collected in the river 9 and 18 km upstream from the station. No radioactivity attributable to the station except ^{3}H was observed in shad that swim past the station to spawn upriver, and none in shellfish collected from the Connecticut River estuary at Old Saybrook.

(4) The benthos of the discharge canal and the Connecticut River from Haddam Island State Park to Salmon Cove was monitored with an underwater NaI(Tl) probe connected to a portable gamma-ray spectrometry system. Sediment samples, mostly bottom sand, were collected by a diver and with dredges dropped from a boat where the probe showed radionuclides from the station, and these were analyzed for photon-emitting radionuclides and ^{90}Sr. The radionuclides ^{54}Mn, ^{57}Co, ^{58}Co, ^{60}Co, ^{134}Cs, and ^{137}Cs from the station were found at the mouth of the discharge canal and at a few nearby locations along the east bank of the Connecticut River.

The utility's contractor for environmental surveillance and the Connecticut Department of Environmental Protection found gross activity attributed to the station in water and in sediment samples at the mouth of the discharge canal in 1970 and 1971.(1,2) Gross-beta activity in excess of background had also been observed in plankton collected from the mouth of the discharge canal, but not in fin-fish or shellfish. No specific radionuclides were reported but ^{54}Mn, ^{58}Co, and ^{60}Co were believed to be in some samples.(2)

For comparison, at the Indian Point 1 PWR low levels of ^{24}Na, ^{54}Mn, and ^{131}I were observed in discharge water with an immersed NaI(Tl) detector;(3) and ^{54}Mn, ^{58}Co, ^{60}Co, ^{134}Cs, and ^{137}Cs were in sediment, aquatic vegetation, and fish below the outfall.(4) At the Yankee-Rowe PWR, ^{3}H was found in effluent coolant-canal water during waste discharges; ^{54}Mn, ^{58}Co, and ^{60}Co were in aquatic vegetation; and ^{54}Mn, ^{60}Co, ^{90}Sr, ^{125}Sb, and ^{137}Cs, in sediment.(5)

Extensive and detailed ecological studies have been conducted on the discharge canal and the Connecticut River since 1965 by The Essex Marine Laboratory to determine the effect of thermal pollution on all phases of aquatic life.(6) These studies did not include radiological measurements, but aided considerably the ones described in this section.

5.1.2 *Connecticut River hydrology.* The Haddam Neck station is located on the Connecticut River approximately 29 km* by river above the mouth of the

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*1 mile = 1.61 km; 1 cubic foot per second (cfs) = 28.3 liters/s

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river in Long Island Sound. The station discharges its wastes into a circulating coolant water canal that empties into the Connecticut River 1.7 km below the station and 1.2 km above the mouth of the Salmon River (see Figure 5.1).

While the site experiences a semidiurnal reversal of tidal flow, saline water from Long Island Sound only extends to 3.2 km south of the station.(7) The results of water analyses for stable substances are shown in Table 5.1. The average flow of fresh water in the river is approximately 18,000 cfs , with average monthly flows as low as 2,000 cfs and as high as 70,000 cfs. Because of the tide, the net minimum daily average flow is 15,000 cfs, and the tidal range in the river at the site is about 1 meter.(7)

5.2 Tritium in River Water

5.2.1 *Sampling and analysis.* Water was collected to measure tritium concentrations beyond the point of release during and after the release of waste liquid from the test tanks (see Section 4.2.1). On Sept. 15, 1970, and March 15, 1971, samples were taken at the locations and times listed in Table 5.2. At each location, water was collected from the surface in 50-ml portions generally at three points across the river.

Table 5.1

Concentration of Stable Substances in Connecticut River Water

	Concentration, µg/liter	
Substance	**Discharge canal at weir, Haddam Neck, 3/15/71**	**Maromas power plant, Middletown, 3/1/71**
Calcium + Magnesium (as $CaCO_3$)	4.2×10^4	4.5×10^4
Iron	98	70
Boron	94	89
Strontium	53	47
Barium	39	29
Zinc	21	19
Copper	7	5
Aluminum	<17	<17
Arsenic	<43	<43
Beryllium	< 1	< 1
Cadmium	< 9	< 9
Chromium	< 4	< 4
Cobalt	< 9	< 9
Lead	<17	<17
Manganese	< 4	< 4
Molybdenum	<17	<17
Nickel	< 9	< 9
Phosphorus	<43	<43
Silver	< 1	< 1
Vanadium	<17	<17

Notes:

1. We thank John F. Kopp, NERC-Cincinnati for these analyses.
2. Concentrations for the stable elements were measured by emission spectrographic analysis, except that calcium + magnesium were determined by complexometric titration.
3. Concentrations of Ca and Mg in the Connecticut River at Haddam were 12 and 2.0 mg/l, respectively, based on averages of 12 monthly analyses beginning Oct. 1971. We thank W. H. Oldaker, Needham Heights Laboratory, EPA, Boston, Mass., for these values.
4. The Maromas power plant is 17 km upstream from the Haddam Neck station.

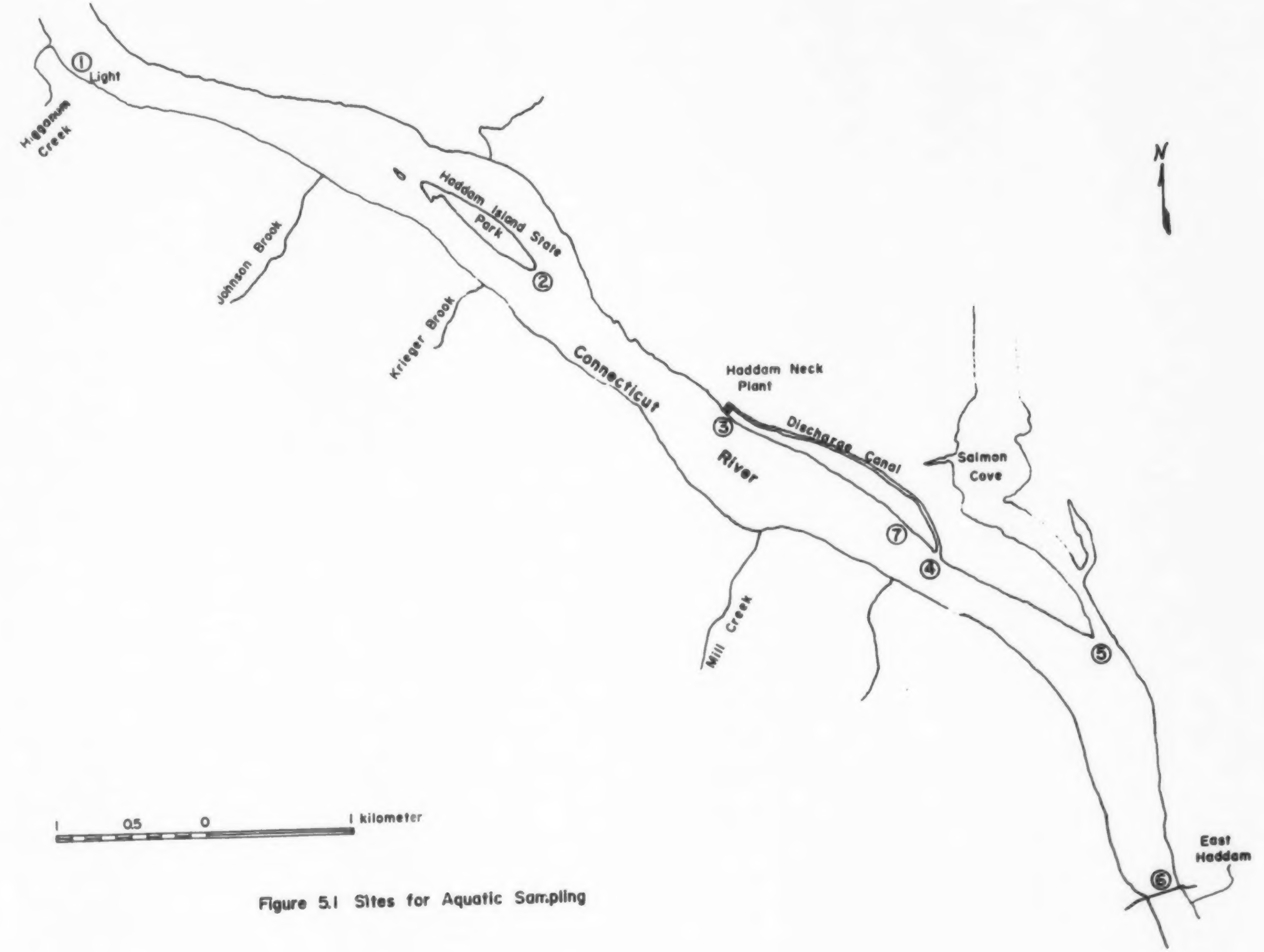
N
① Light
Higganum Creek
Johnson Brook
Haddam Island State Park
②
Krieger Brook
Connecticut River
Haddam Neck Plant
③
Discharge Canal
Salmon Cove
⑦
④
Mill Creek
⑤
East Haddam
⑥
1 0.5 0 1 kilometer

Figure 5.1 Sites for Aquatic Sampling

Table 5.2

Tritium Concentrations at Sampling Points in Connecticut River, pCi/ml

	Sept. 15, 1970			March 15 1971		
Sampling Point	**1300 hrs**	**1500 hrs**	**1700 hrs**	**1300 hrs**	**1500 hrs**	**1700 hrs**
Higganum Light (1)*	NS†	<0.6	NS	NS	NS	NS
Haddam Island (2)	1.3	NS	<0.6	<0.9	1.0	1.0
Water intake (3)	NS	NS	<0.6	1.0	2.8	1.0
0.3 km above canal mouth (7)	4.0	5.8	1.9	NS	NS	NS
Canal mouth at log barrier (4)	3.7	5.6	2.9	63	63	64
Mouth of Salmon River (5)	8.4	3.9	6.1	1.4	1.2	2.5
Haddam Bridge (6)	10.6	10.7	5.9	NS	NS	NS

*Numbers in parenthesis refer to the locations in Figure 5.1

†NS - not sampled

The water samples were prepared for tritium analysis by distilling at least 10 ml of water to separate tritium from nonvolatile radionuclides. The distilled water was then mixed with scintillating solution to measure the tritium in a liquid-scintillation counter. The energy-response settings of the counter were adjusted to optimize detection of the low-energy beta particles of ^{3}H. For routine analysis, the minimum detectable concentration was 0.6 pCi/ml.*

5.2.2 *Results and discussion.* The ^{3}H concentrations in the discharge canal on September 15, 1970, and on March 15, 1971, computed from analyses of station effluents, were 24 and 81 pCi/ml, respectively (Table 4.6 and 4.7). The measurement results on September 15, listed in Table 5.2, were lower than predicted at the mouth of the discharge canal, but concentrations were higher at Haddam Bridge, after at least some additional dilution. The pattern of concentrations (see also Table 4.6) suggests that the discharge rate was lower than the value given in Section 4.2.1 during part of the release period. Water from the canal appears to have moved on the surface either up- or down-stream in response to tidal conditions without very much dilution. On March 15, ^{3}H concentrations in the canal mouth were reasonably consistent with discharge values. The ^{3}H up- and down-stream in the Connecticut River was much diluted, although measurable.

The observations suggest that ^{3}H in reactor system liquid wastes, although at low concentrations relative to the discharge limits, can be used as a tracer for flow and dispersion studies. Studies for this purpose would require much more sampling as a function of distance, cross-section, and time than was possible in this study.

5.3 Other Radionuclides in River Water

5.3.1 *Gross activity measurements.* Measured radionuclide concentrations in canal water and Connecticut River water at the station water intake are reported in Tables 4.6 and 4.7. Upper concentration limits of these radionuclides in the Connecticut River at the other locations listed in Table 5.2 can be inferred from their measured values in Tables 4.6 and 4.7 relative to those of ^{3}H.

The utility's contractor for environmental surveillance and the Connecticut Department of Environmental Protection measured the following gross activity values in Connecticut River water:(1,2)

	Gross beta activity, pCi/liter		
Location*	**1969(1)**	**1970(1)**	**1971(2)**
Higganum Light (1)†	**3.8**	**4.3**	**3.4**
Discharge canal (4)	**6.8**	**9.9**	**3.9**
Salmon River (5)	**4.0**	**3.9**	**4.3**
HaddamBridge (6)	**4.0**	**5.3**	**4.1**

***Numbers in parentheses refer to map locations in Figure 5.1.**

†Control (background) sample relative to the Haddam Neck plant.

The gross beta activity was usually above the background level at the mouth of the discharge canal, and occasionally so at the downstream locations. The highest concentration, at location #4 in 1970, was probably due to test tank discharges during sample collection. The annual average gross beta activity in the discharge canal, based on analyses in liquid waste by the station operator (see Section 4.1.2), was 9 pCi/liter in 1970 and 1971. Concentrations beyond the mouth of the discharge canal would be less to the extent of dilution in the Connecticut River. Average annual gross alpha activities were below the minimum detection limit, at the 3-sigma level, of 1 pCi/liter or less.[1]

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*We thank R. Lieberman, EERF, ORP-EPA, for analyzing some of the samples.

5.3.2 *Average radionuclide concentrations in the discharge canal.* Because concentrations of radionuclides discharged by the plant were in many cases near or below minimum detectable levels in canal and river water, average concentrations in canal water were calculated from concentrations in samples of liquids before discharge. Below are the average radionuclide concentrations calculated to be in the discharge canal during the study period:

Radionuclide	Calculated radionuclide concentration, pCi/liter: from values reported by station	from measured in-plant samples, this study
12.3 -yr ^{3}H	9,700	11,000
5730 -yr ^{14}C	NR	0.04
15 -h ^{24}Na	NR	0.03
14.3 -d ^{32}P	NR	0.01
27.7 -d ^{51}Cr	NR	0.007
313 -d ^{54}Mn	0.42	0.1
2.7 -yr ^{55}Fe	NR	0.6
44.6 -d ^{59}Fe	NR	0.01
270 -d ^{57}Co	NR	0.008
71.3 -d ^{58}Co	3.6	0.6
5.26-yr ^{60}Co	0.56	0.3
28.5 -yr ^{90}Sr	NR	0.0007
65 -d ^{95}Zr	NR	0.001
35.1 -d ^{95}Nb	NR	0.004
66.2 -h ^{99}Mo	NR	0.04
253 -d ^{110m}Ag	NR	0.004
8.06-d ^{131}I	2.0	8
20.9 -h ^{133}I	0.94	3
6.7 -h ^{135}I	NR	1
2.07-yr ^{134}Cs	NR	0.1
13 -d ^{136}Cs	NR	0.03
30 -yr ^{137}Cs	0.50	0.1

Notes: approximately 23 Ci of ^{133}Xe and 0.10 Ci of ^{135}Xe were discharged annually in the water, but aeration would be expected to expel these nuclides; ^{89}Sr was not detected in waste liquids.

NR – Not Reported.

The average concentrations listed in the first data column are based on discharges in 1970 and 1971 and the total available dilution reported by the station (see Section 4.1.2). The average concentrations listed in the second data column are the summed annual discharges listed in Tables 4.4 and 4.5, divided by the total dilution volume in the canal of 7.3 x 10^{11} liters/yr. The values in the first data column, based on monitoring all discharges, should be superior to those based on the occasional samples in this study. Actual concentrations would fluctuate about these averages because reactor system wastes are discharged periodically. Concentrations in the Connecticut River, after the canal water had been thoroughly mixed in it, would be lower, on the average, by a factor of 25.

5.4 Radionuclides in Vegetation, Plankton, and Algae

5.4.1 *Sampling and analysis.* Four samples of aquatic plants were collected from the Connecticut River on September 15, 1970—one sample each of *Vallisneria americana* and *Potamogeton sp.* 8 km upstream from the mouth of the discharge canal at location 1 (see Figure 5.1), and two samples of *V. americana* collected at the boat dock near the plant water intake (location 3, Fig. 5.1).* Samples at location 1 were considered as controls. Neither *V. americana* nor *Potamogeton* was observed growing in the discharge canal.

Two samples of plankton, including both phyto- and zoo- plankton with some detritus, were collected on September 15, 1970.* One sample was collected at the mouth of the discharge canal by hanging a 1-m-diameter plankton net from the log barrier for 12.2 hours. Based on a flow rate of 0.33 m/s in the canal, the sampled water volume was estimated to be 11,500 m³. The control sample was collected by towing a 0.5-m-diameter plankton net near the mouth of Higganum Creek (location 1, Figure 5.1). The volume of sampled water, measured with a flow meter, was 75 m³. The species of collected plankton were not identified.

Two algae samples were collected from the wooden barrier across the mouth of the circulating coolant discharge canal. The first was obtained on September 15, 1970; the second, on March 2, 1971. The algae samples consisted mostly of blue-green algae *(Lyngbya, Oscillatoria, Phormidium)* with some green algae *(Cosmarium)* and diatoms *(Nitzschia, Melosira).*** Attempts were made to locate algae upriver from the station, but no adequate quantities of growth were found.

Gamma-ray analyses were initially performed on the fresh samples to measure any photon-emitting radionuclides that might be volatile during ashing, particularly radioiodine and radioruthenium. The samples were then dried at 100° C and ashed at 400° C.

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*We thank R. R. Massengill and associates, Essex Marine Laboratory, for collecting and identifying these samples.

**We thank Dr. William Brungs and associates, Newtown Fish Toxicology Laboratory, EPA, for identifying the algae samples.

Samples were analyzed for photon emitters by spectrometry with an 11-cm^3 Ge(Li) detector (see Figure 5.2), and with a NaI(Tl) gamma-ray coincidence/anticoincidence system. Radiochemical analyses were performed to measure ^{89}Sr, ^{90}Sr and ^{32}P;(9) ^{3}H and ^{14}C were determined by treating samples in a combustion train, collecting H_2O and CO_2, and measuring the radioactivity with liquid scintillation and gas counting techniques.† Stable strontium and calcium were measured by an atomic absorption spectrophotometer.

5.4.2 *Results and discussion.* The concentrations of radionuclides measured in the four samples of aquatic plants are listed in Table 5.3. The control samples contained only ^{55}Fe, ^{95}Nb, ^{137}Cs, and ^{144}Ce from atmospheric fallout. Both samples collected near the station contained measurable quantities of radionuclides discharged by the station; these included ^{32}P, ^{54}Mn, ^{58}Co, ^{60}Co, ^{90}Sr, and ^{131}I. The fallout radionuclides ^{95}Nb, ^{137}Cs, and ^{144}Ce in these samples were at somewhat lower concentrations than in the controls. Radionuclides not detected in any of these samples at the 3-sigma detection limit of 25 pCi/kg were ^{57}Co, ^{89}Sr, ^{95}Zr, and ^{134}Cs.

The *V. americana* found floating at flood tide near the boat dock contained higher radionuclide levels than the plants that grew at the sampling location. It is possible that the floating plants had grown nearer the mouth of the discharge canal, were uprooted by carp or catfish searching for food, and forced upriver by the rising tide.(10) The detection of discharged radionuclides in plants growing upstream from the mouth of the discharge canal is in accord with observations of the tidal movement of water upstream, past the station water intake, in Sections 4.3.4 and 5.2.2.

The plankton from the discharge canal contained a number of radionuclides indicative of station wastes that were not detected in the control sample—^{54}Mn, ^{58}Co, ^{60}Co, ^{95}Nb, ^{131}I, and ^{134}Cs (see Table 5.4). Observed in both the canal and control samples were ^{137}Cs and ^{144}Ce from fallout; ^{95}Nb, which may be from fallout, was only in the canal sample. Although ^{14}C was not measured in the control plankton sample, the specific activity of ^{14}C in the canal sample is higher than the usual specific activity of 6.1 pCi/g carbon in contemporary samples. The plankton is exposed to plant discharges for a relative short time—about 80 min during transit in the 1.8-km canal, hence equilibrium may not be attained between the radionuclide content of the plankton and the water in the canal. The results illustrate the strong tendency of plankton, a principal food for some species of fish, to concentrate certain radionuclides.

Algae samples from the barrier at the mouth of the discharge canal contained relatively high concentrations of the following radionuclides discharged from the station: ^{54}Mn, ^{55}Fe, ^{58}Co, ^{60}Co, ^{89}Sr, ^{131}I, and ^{134}Cs (see Table 5.4). The higher levels observed in the earlier sample may have resulted from higher radionuclide concentrations in effluent, faster algae growth, a possibly greater uptake of nuclides during the higher temperatures of summer (35–40° C *vs.* 12–13° C in winter),(11) the age of the algae,(12) or a longer period of growth. The first sample had grown in the discharge canal for an undetermined but probably lengthy period, while the later sample had grown only during the 5-month interval since the first collection. Some or all of the ^{55}Fe, ^{90}Sr, ^{95}Zr, ^{95}Nb, ^{137}Cs, and ^{144}Ce may be from fallout. In the first sample only, the ^{14}C specific activity exceeded the normal value. Tritium, measured only in the later sample, was at a higher concentration than the approximately 500 pCi/liter water of combustion usually observed in environmental samples at the time.(13) The observed ^{3}H level in the algae sample is consistent with the average ^{3}H concentration in canal water of 10 nCi/l calculated in Section 5.3.2.

Unlike the plants and plankton, the algae samples were submerged in the canal water continuously during their entire growing period. Hence, radionuclide concentration factors (CF) were calculated for the first algae sample based on the concentrations measured in the sample and the average water concentrations derived in Section 5.3.2 in this study. The CF (pCi/kg algae per pCi/kg water) for this mixture of green and blue-green algae are:

^{14}C = 1 x 10^4
^{54}Mn = 1 x 10^5
^{57}Co = 5 x 10^4
^{58}Co = 6 x 10^4
^{60}Co = 7 x 10^4
^{131}I = 1 x 10^3
^{134}Cs = 1 x 10^4
^{137}Cs = 1 x 10^4

To compute these CF values the following background concentrations for radionuclides in fallout were added to the concentrations in water from station operation given in Section 5.3.2: 0.1 pCi ^{137}Cs/liter and 0.067 pCi ^{14}C/liter. The latter is based on the assumption of 11 ppm carbon in fresh water with a specific activity of 6.1 pCi ^{14}C/g carbon.(14)

All of these concentration factors are between 10^3 and 10^5. The three cobalt isotopes showed similar values, as did the two cesium isotopes. The large concentration factors may make algae an important

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†We thank E. J. Troianello, EPA, Winchester, Massachusetts, for the ^{14}C and ^{3}H analyses of all aquatic samples.

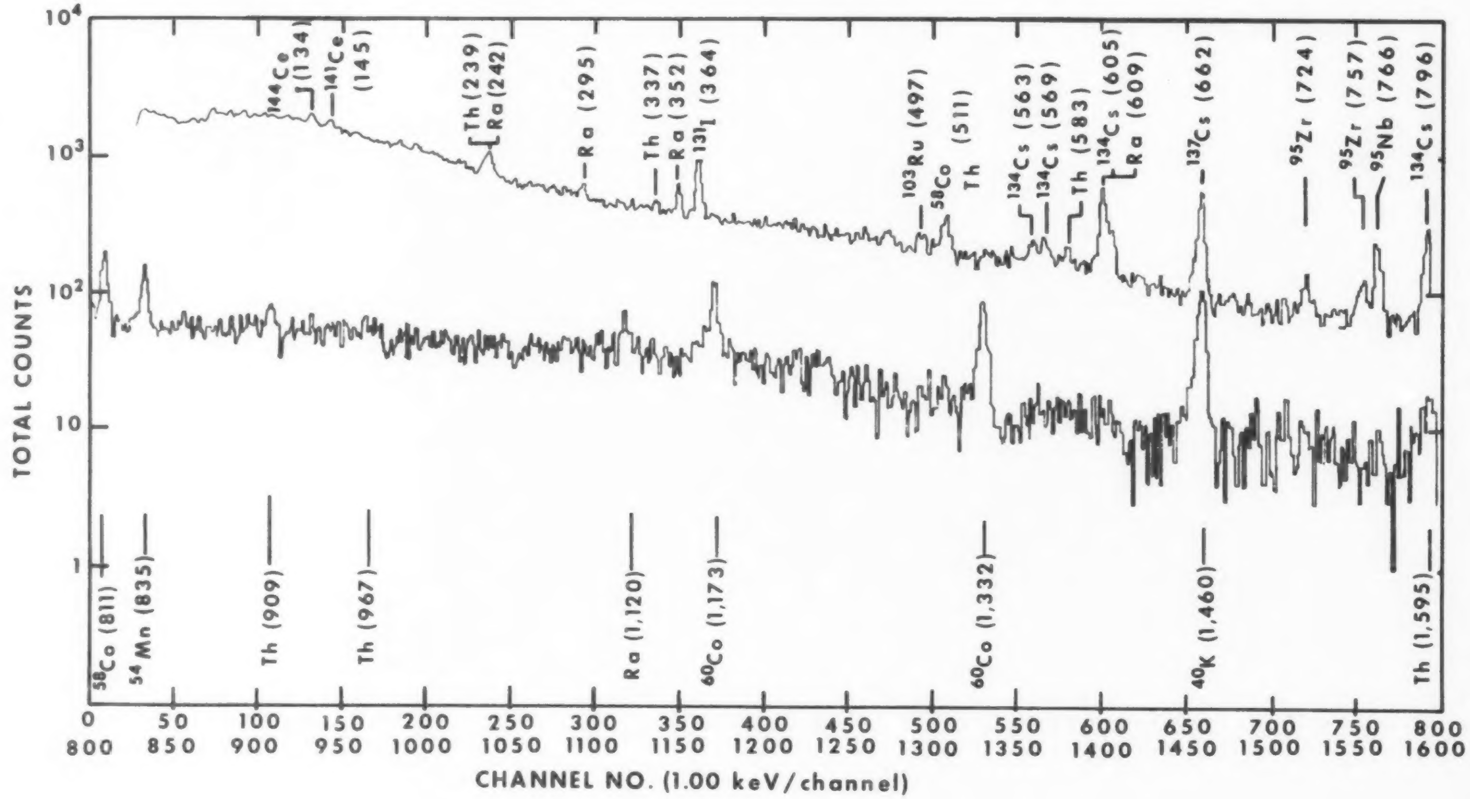

Figure 5.2 Gamma-ray spectrum of algae
Detector: Ge(Li), 10.4 $cm^2 \times$ 11 mm, trapezoidal.
Sample: 467 gm (400 cc) fresh wt., collected March 2, 1971 from log barrier at mouth of circulating coolant discharge canal.
Count: March 10-11, 1971 (1460 min., background not subtracted); Ra and Th refer to ^{226}Ra and ^{232}Th plus progeny.

Table 5.3

Radionuclide (pCi/kg)* and Stable Ion (g/kg)* Concentrations in Aquatic Plants

	near water intake (location 3)		Higganum Light (location 1)	
Nuclide	*V. americana*†	*V. americana*	*Potamogeton sp.*	*V. americana*
^{32}P	31 ± 17	< 15	< 15	< 10
^{54}Mn	140 ± 22	49 ± 8	< 10	< 20
^{55}Fe	NA	600 ± 200	370 ± 60	< 250
^{58}Co	120 ± 18	72 ± 9	< 10	< 20
^{60}Co	25 ± 9	< 20	< 20	< 30
^{90}Sr	26 ± 9	< 10	< 10	< 10
^{95}Nb	85 ± 20	45 ± 9	120 ± 40	85 ± 30
^{131}I	250 ± 24	160 ± 90	< 20	< 25
^{137}Cs	14 ± 7	16 ± 10	31 ± 11	30 ± 16
^{144}Ce	150 ± 50	110 ± 26	270 ± 40	230 ± 60
K	0.36	0.36	0.32	0.40
Ca	0.47	0.53	0.40	0.72
Sr	0.0032	0.0029	0.0021	0.0043
ash wt./wet wt.	0.017	0.015	0.010	0.019

*kg wet weight

†found floating at flood tide; all other samples were growing at the sampling location

Notes:

± values are 2-sigma and < values are 3-sigma counting error.

NA – not analyzed

Table 5.4

Radionuclide (pCi/kg)* and Stable Ion (g/kg)* Concentration in Plankton and Algae

	Plankton (Sept. 15, 1970)		Algae at canal mouth (location 4)	
Nuclide	Canal mouth (location 4)	Higganum Light (location 1)	Sept. 15, 1970	March 2, 1971
^{3}H	NA	NA	NA	7,400 ± 800†
^{14}C	10.4 ± 0.3(71)**	NA	30.0 ± 1.0(390)**	6.4 ± 0.5(5)**
^{54}Mn	1,200 ± 150	< 150	11,000 ± 1,100	560 ± 50
^{55}Fe	NA	NA	41,000 ± 4,000	7,000 ± 2,000
^{57}Co	< 40	< 40	370 ± 150	< 30
^{58}Co	3,700 ± 200	< 120	39,000 ± 1,500	730 ± 60
^{60}Co	1,600 ± 200	< 200	21,000 ± 1,400	640 ± 80
^{90}Sr	< 30	< 50	42 ± 5	28 ± 4
^{95}Zr	< 90	< 190	1,800 ± 1,200	970 ± 100
^{95}Nb	560 ± 160	< 120	1,300 ± 900	870 ± 70
^{131}I	8,900 ± 600	< 100	11,000 ± 1,100	2,300 ± 80
^{134}Cs	570 ± 150	< 150	1,400 ± 700	1,200 ± 70
^{137}Cs	420 ± 110	340 ± 110	2,000 ± 800	1,900 ± 80
^{144}Ce	820 ± 300	1,100 ± 600	4,600 ± 800	3,800 ± 180
K	2.1	2.3	0.6	0.8
Ca	0.53	0.50	1.23	1.10
Sr	0.0044	0.0040	0.0044	0.0048
Fe	NA	NA	15	11
ash wt./ wet wt.	0.047	0.079	0.123	0.22

*kg wet weight

**^{14}C concentrations are in pCi/g carbon, and the values in parentheses are the percent excess ^{14}C (natural concentration = 6.1 pCi/g C); algae samples contained 3.9% carbon by weight.

†Concentration in units of pCi/liter of water of combustion (measured 78% of fresh sample weight)

NA- not analyzed

link in the chain of radionuclide transfer from water through fish to man. Relative ease of analysis makes them useful indicators of radionuclides in the aqueous environment.

5.5 Radionuclides in Fish

5.5.1 *Introduction.* The only commercial fish in the Connecticut River is shad in the lower 43 km (27 miles) of the river, but the river is used extensively for sport fishing.(7) Popular fishing locations on the river in the vicinity of Haddam Neck are near the mouth of the Salmon River (Figure 5.1, location 5) 1.2 km downstream from the discharge canal, and the warm water near the mouth of the discharge canal. At the time of the study, fishing was not permitted beyond the barrier at the mouth of the canal. More recently, the station operator has provided fishermen access to the banks of the canal.(10)

The discharge canal offers an unusual habitat for studying radionuclide uptake by fish.(15,16) Certain species of fish are attracted to and remain active in the heated effluent canal, whose water is as much as 12.4° C above the ambient temperature during the colder months of the year. In the colder river water, the fish become inactive, cease eating, and some may even burrow in the bottom sediments when the water temperature reaches 6–8° C. Occupancy of this "warm-water" habitat results in an abnormal seasonal metabolism as well as feeding and activity behavior. Brown bullheads in a control area (a cove 8 km downstream from the discharge canal, and 3.2 km below any detectable heated water from the plant) all had empty stomachs in December when the water temperature was 2° C; a comparable sample from the canal at this time showed that 30 percent of the fish had fed. In March, when the control area was 4° C, 27 percent of the bullheads had fed, while 51 percent of those in the 13° C canal water were feeding actively.(16) During the spring and early summer months when the water temperature increases, fish begin to leave the canal. The physical condition of the canal fish is somewhat impaired because: (1) high population densities in the canal result in overcrowding and increased competition for food (an average of 30,000 fish was estimated to be in residence during the winter),(16) (2) metabolic rates of the canal fish in winter are higher than for fish in waters at ambient temperature, and (3) more energy is required for fish to maintain themselves in the canal at a flow rate of 0.3 m/s.(15) Hence, the uptake of radionuclides by these fish may be atypical.

The main foods normally consumed by the various fish species in the canal are:

Carp (bottom scavengers)
Juvenile – live organisms; crustaceans, larvae, and worms
Adult – (not discrete in food selection); silt, leaves, roots, sticks, decaying fish, and sometimes insects.
White catfish (selective)
Juvenile – small crustaceans (copopods, cladocearns), fly pupae, and larvae.
Adult – small fish, worms, larvae, plankton, and some vegetation. Catfish will feed on the bottom when food is not available in water. They do not actively feed in winter.
Bullhead (scavengers)
Juvenile – crustaceans (copopods, cladocearns), insect larvae and vegetation; they feed about 50 percent of the time in the water column and 50 percent of the time on the bottom.
Adult – small fish, detritus, insect larvae, vegetation, roots, stems and decayed material on bottom.

Although these foods were the basic diet of the fish, a wide variety of small amounts of other matter was observed in the stomachs of the fish.(10)

Because commercial fishing for the American shad *(Alosa sapidissima)* could be affected by the aqueous discharges from the nuclear power station, the fish have been a major subject for study by the Essex Marine Laboratory. Shad is a marine fish that enters the Connecticut River estuary in April, swims upstream past the nuclear power station, and spawns far upriver beyond Hartford, Connecticut. After spawning, the shad return to the ocean, followed in late summer or early autumn by the juvenile fish. The adult shad do not eat while in the river, but the juvenile fish will eat on their passage to the ocean. During their migration from the mouth of the river to the station, the shad are exposed to aqueous discharges from the station for approximately 1 day, but are in the immediate neighborhood of the station—say 10 km downstream—for only approximately 6 hours.(10)

5.5.2 *Collection and analysis.* Fresh-water fish were collected from the discharge canal on two occasions by trawling. At the same time, background fish were collected from the Connecticut River at Higganum and Middletown, about 9 and 18 km above the discharge canal, respectively.* The collected fish are listed in Table 5.5. The fish species from the discharge canal—white catfish, brown bullhead, and carp—composed more than 95 percent of the fish species population of the canal in winter.(17) Control fish for the March sampling period were obtained from the intake screens at the fossil-fueled Maromas power plant.

Shad that had migrated past the station were collected on two occasions about 30 km upriver at Rockyhill, Connecticut. Control shad were taken from the estuary.

Samples were frozen immediately after collection and returned to the laboratory on dry ice. For analysis, the fish were thawed, weighed, and dissected for the following tissues that were expected to concentrate the radionuclides of interest:

muscle	**– ^{134}Cs and ^{137}Cs analysis**
bone	**– ^{89}Sr and ^{90}Sr analysis**
kidney & liver	**– ^{58}Co, ^{60}Co, and ^{55}Fe analyses**
thyroid in heart area and head kidney†	**– ^{131}I analysis**
stomach & gut	**– gamma-ray spectral analysis**

After dissection, the tissues were combined by species of fish for each sampling period. Special care was exercised during dissection to obtain bone-free muscle samples.(5) The separation of muscle from the skeleton was facilitated by cooking in a microwave oven. However, some small bones were retained in the muscle, particularly in the small fish, as will be discussed later.

To measure volatile radionuclides, all soft tissues were analyzed before ashing by gamma-ray spectrometry with a 10 x 10-cm NaI(Tl) detector or an 11-cm^3 Ge(Li) detector. For more sensitive ^{60}Co and ^{131}I measurement, samples of liver-kidney and thyroid were analyzed with a NaI(Tl) gamma-ray coincidence/anti-coincidence system. Tritium and ^{14}C were determined in some muscle and kidney-liver samples by treating them in a combustion train, collecting water and CO_2, and measuring the radioactivity with liquid scintillation and gas counting techniques.**

Bone was ashed at 600° C, and strontium was separated chemically. Radiostrontium was measured by counting total strontium and ^{90}Y.(19) Stable strontium and calcium were determined by an atomic absorption spectrophotometer.

Muscle and gut were dried at 100° C, ashed at 400° C, and then analyzed by gamma-ray spectrometry. Radiochemical analyses were performed to measure ^{90}Sr. The potassium content of the muscle was calculated from the ^{40}K gamma-ray measurement (848 pCi ^{40}K/gm K), and the stable calcium and strontium concentrations were determined by an atomic absorption spectrophotometer.

Liver-kidney samples were ashed and the iron fraction was separated chemically. It was analyzed for ^{55}Fe with an X-ray proportional detector, and for stable iron with an atomic absorption spectrophotometer.

5.5.3 *Results and discussion.* The results of the fish analyses are given in Table 5.6. The concentrations are listed relative to wet weights, and the ash weight/wet weight ratios are given to convert concentration to an ash weight basis if desired.

The average ^{90}Sr concentrations in the bone of background and canal fish were the same (see Table 5.7), as were also the stable strontium concentrations. The average ^{90}Sr concentration of all fish bone is 600 ± 130 (1-sigma) pCi/kg wet weight, 12.1 ± 2.3 pCi/gm Ca, and 7.0 ± 1.6 pCi/mg Sr. These values are about 5 times less than the average concentration of 2800 pCi/kg(5) measured in bone of fish collected in June 1969, from the Sherman and Harriman reservoirs in western Massachusetts. The lower ^{90}Sr concentrations may be a result of the higher calcium content of the river water. At a ^{90}Sr concentration—from fallout—in canal water of 0.6 pCi/liter (see Section 4.3.4) and a calcium concentration in the water of 12 mg/liter (see note 3 to Table 5.1), the average observed ratio is 0.24, as given by:

$$OR_{bone\ water} = (^{90}Sr/Ca)_{bone}/(^{90}Sr/Ca)_{water}$$

This is within the range of 0.1–0.7 in published values.(20-23) For stable strontium relative to calcium, the mean ratio in fish bone was 1.8 ± 0.4 mg Sr/gm Ca; based on this value in bone and the strontium and calcium concentrations in canal water of Table 5.1, the average $OR_{bone/water}$ was 0.4. No ^{89}Sr was detected in any of the fish bone samples at the minimum detectable

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*We thank R. R. Massengill and associates, Essex Marine Laboratory, for collecting the fish samples.

† It is reported that more than 50 percent of the iodine accumulates in the head kidney of some fish species.(18)

**We thank E. J. Troianello, EPA, Winchester, Mass., for the ^{3}H and ^{14}C analyses.

Table 5.5

Fish Collected at Haddam Neck

Date	Location	Fish Species	Number	Average wt., g
Dec. 1, 1970	Conn. R. at Higganum	White catfish *(Ictalurus catus)*	8	500
Dec. 4, 1970	discharge canal	White catfish *(Ictalurus catus)*	10	850
		Carp *(Cyprinus carpio)*	6	300
Mar. 2, 1971	Conn. R. at Middletown	Crappie *(Pomoxis sp.)*	3	
		Yellow perch *(Perca flavescens)*	4	250
		Brown bullhead *(Ictalurus nebulosus)*	2	
	discharge canal	White catfish *(Ictalurus catus)*	4	850
		Carp *(Cyprinus carpio)*	2	5,300
		Brown bullhead *(Ictalurus nebulosus)*	5	600
Apr. 15, 1971	Conn. R. estuary	Shad *(Alosa sapidissima)*	2	1,900
Apr. 26, 1971	Conn. R. at Rockyhill	Shad *(Alosa sapidissima)*	2	1,500
June 1, 1972	Conn. R. estuary	Shad *(Alosa sapidissima)*	2	3,000
	Conn. R. at Rockyhill	Shad *(Alosa sapidissima)*	2	1,600

Table 5.6

Radionuclide (pCi/kg)* and Stable Ion (g/kg)* Concentrations in Fresh Water Fish

Sample	Nuclide	December 1–4, 1970			March 2, 1971			
		Carp	Catfish	Catfish Background	Carp	Catfish	Bullhead	Mixed Fish Background
Bone	Ash wt./wet wt.	0.17	0.15	0.14	0.12	0.13	0.13	0.15
	^{90}Sr	840 ± 30	610 ± 40	620 ± 40	440 ± 40	460 ± 40	650 ± 30	580 ± 50
	Sr	0.083	0.090	0.091	0.059	0.091	0.099	0.105
	Ca	51	50	51	45	47	49	53
Muscle	Ash wt./wet wt.	0.019	0.018	0.015	0.013	0.019	0.015	0.013
	^{3}H	NA	NA	NA	4500 ± 400	2400 ± 300	5700 ± 400	600 ± 160
	^{14}C(pCi/g C)**	NA	NA	NA	7.0 ± 0.3	5.9 ± 0.4	8.7 ± 0.3	6.3 ± 0.3
	^{90}Sr	29 ± 2	34 ± 2	15 ± 2	3.5 ± 0.4	30 ± 1	20 ± 1	NA
	^{134}Cs	140 ± 8	57 ± 6	< 10	< 30	< 30	43 ± 10	< 30
	^{137}Cs	320 ± 8	190 ± 6	120 ± 8	100 ± 10	150 ± 10	190 ± 10	49 ± 9
	Sr	0.0029	0.0032	0.0017	0.00057	0.0047	0.0021	0.0014
	Ca	2.13	2.41	0.92	0.45	2.97	1.50	0.78
	K	3.76	3.46	3.76	3.72	3.70	3.60	3.36
	Fe	NA	NA	NA	0.021	0.0093	0.0073	NA
Liver plus kidney	H	1300 ± 400	700 ± 400	NA	NA	NA	NA	NA
	^{14}C(pCi/g C)	12.5 ± 0.3	8.0 ± 0.1	NA	NA	NA	NA	NA
	^{55}Fe	NA	NA	NA	3500 ± 400	600 ± 200	1300 ± 200	4300 ± 300
	^{131}I†	1800 ± 130	< 30	< 50	460 ± 20	200 ± 20	250 ± 20	< 160
	Fe	NA	NA	NA	0.194	0.091	0.094	NA
Gut	^{58}Co	290 ± 60	< 50	< 70	< 20	< 30	< 30	< 80
	^{60}Co	140 ± 10	< 40	< 30	< 20	60 ± 16	80 ± 10	< 50
	^{131}I†	< 60	< 80	< 180	220 ± 30	670 ± 60	1500 ± 40	< 70
	^{134}Cs	200 ± 50	110 ± 20	< 50	< 20	80 ± 18	130 ± 20	< 60
	^{137}Cs	380 ± 60	280 ± 50	74 ± 10	100 ± 50	190 ± 20	260 ± 10	90 ± 40

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* kg wet weight

**11.0 ± 0.8 percent carbon by weight (average measured value)

† Confirmed by decay measurements

NA – not analyzed

Notes: ± values are 2-sigma and < values are 3-sigma counting error.

Table 5.7

Average Radionuclide Concentrations in Bone and Muscle in Fresh Water Fish, pCi/kg*

Radionuclide	Canal fish	Background fish
	Bone	
^{90}Sr	600 ± 160	600 ± 30
	Muscle	
^{3}H	2900 ± 2100	600
^{14}C	900 ± 300	670
^{90}Sr	23 ± 12	15
^{134}Cs	50 ± 50**	< 20
^{137}Cs	190 ± 80	80 ± 40

*kg wet weight; ± values are 1-sigma of individual concentrations.

**Values below the detection limit were averaged as one-half of this limit.

Notes:

1. Concentrations of ^{3}H and ^{14}C measured in kidney plus liver samples were averaged with concentrations in muscle, on the assumption that the two radionuclides were uniformly distributed in these tissues.
2. The concentration of ^{14}C was computed from the specific activity on the basis of 110 g C per kg muscle.
3. The concentrations of ^{3}H and ^{90}Sr in background fish are from single samples; the specific activity of ^{14}C in background fish was taken to be 6.1 pCi/g C.

concentration of 60 pCi/kg fresh weight (3-sigma counting error).

Measured values of stable strontium and ^{90}Sr in the muscle are believed to be due, in part, to small bones analyzed with the meat, although they would normally be removed when eaten. Note that the large carp collected on March 2, 1971, for which meat and bone could be separated efficiently, contained very little calcium, strontium, and ^{90}Sr in muscle. A previously reported[9] ratio of strontium concentration in muscle to that in bone of 0.01 is consistent with the results of stable strontium and ^{90}Sr in this carp sample. This factor, applied to the average concentration of ^{90}Sr in Table 5.7 of 600 pCi/kg bone, yields an average of 6 pCi/kg muscle, which is considered more applicable than the average measured value for fish muscle in Table 5.7.

The concentrations of ^{137}Cs in fish muscle were higher for fish collected from the canal than for those collected upriver in each of the two sets of samples, although the 1-sigma values of the averages overlap (see Table 5.7). The average ^{137}Cs concentration relative to potassium was 52 ± 21 pCi/g K in the flesh of the canal fish, and 22 ± 12 pCi/g K in the two control samples. The average ^{137}Cs concentration in the canal fish was not high in comparison to some other values in fish from waters that do not contain discharges from nuclear power reactors,(24–26) and less than in fish from the Sherman and Harriman reservoirs(5). The presence of ^{134}Cs in some canal fish samples but not in background fish (see Table 5.6), also suggests that part of the ^{137}Cs in fish from the canal is due to effluents from the station. The average $^{134}Cs/^{137}Cs$ ratio in fish flesh was 0.3 ± 0.1, about 1000-fold greater than in fallout.(27) The ratio in water was approximately 0.5, based on average ^{134}Cs and ^{137}Cs concentrations of 0.1 pCi/liter due to effluents (see Section 5.3.2) and the ^{137}Cs concentration in water of approximately 0.1 pCi/liter from fallout (see Section 4.3.4).

Cesium-134 was also detected in the stomach and gut of all but one of the canal fish samples. The average $^{134}Cs/^{137}Cs$ ratio was 0.5 ± 0.1 in these samples. Due to the unusual eating habits of the fish in winter discussed above, many (60–75 percent) of the stomachs and guts of these fish were empty. Therefore, the detection of radiocesium in these samples may reflect to a large degree the concentration in the stomach and gut wall, rather than in the food. The amount of stomach contents was too small for analysis.

The specific activity of ^{14}C in muscle was higher in two of the three canal fish collected on March 2 than in the background sample. Values for the background sample and the third catfish from the canal agreed within the uncertainty of measurement with the usual specific activity. Specific activity values above this value were also measured in the liver and kidney (the only tissue remaining for analysis) of fish caught in the canal on December 1, 1970.

The ^{3}H concentration in three muscle samples and in one of the two kidney plus liver samples was greater than in the background sample. The excess of ^{3}H concentration in canal fish over background fish of 2,300 pCi/kg (see Table 5.7), at a water content of 0.8 liter/kg in fish corresponds to a ^{3}H concentration of 2,900 pCi/liter, approximately one-third of the annual average value from station discharges in canal water computed in Section 5.3.2. The concentration of ^{3}H in the background fish, at a water content of 0.8 liter/kg, was 700 pCi/liter, consistent with background values for river water reported in Section 4.3.4.

No ^{131}I, ^{60}Co, ^{58}Co, or ^{55}Fe was detected in muscle. The minimum detectable levels were approximately 20 pCi ^{60}Co/kg, 40 pCi ^{58}Co/kg, 80 pCi ^{131}I/kg, and 100 pCi ^{55}Fe/kg (10 pCi ^{55}Fe/mg iron).

Radiocobalt was detected in the gut, ^{131}I, in gut and in kidney plus liver samples, and ^{55}Fe, in kidney plus liver samples. The highest concentration of ^{55}Fe was in the background sample, suggesting that this radionuclide originated in fallout. The average ^{55}Fe concentration was 2,400 ± 1,700 pCi/kg wet weight or 13 ± 5 pCi/mg iron. This specific activity falls within the range of 3–50 pCi/mg iron in liver of fresh water fish collected in Finland during 1955.(28) That the gut, kidney, and liver of fish contain these radionuclides has no health implications because they are not eaten by man, but these tissues may, within limits, be useful indicators for radionuclides in edible tissue. For example, the ^{55}Fe specific activity of 13 ± 5 pCi/mg iron may be a better value for muscle than the value from direct measurement of < 10 pCi/mg iron. The specific activity in muscle may be lower, however, if equilibrium has not been attained.

In shad, only ^{3}H and ^{137}Cs were higher upstream from Haddam Neck than in the estuary (see Table 5.8). The higher ^{137}Cs concentrations may have resulted from exposure to ^{137}Cs in fresh water with its much

Table 5.8

Radionuclide (pCi/kg) and Stable Ion (g/kg) Concentrations in Shad

		April 15 + 26, 1971		June 1, 1972	
Sample	**Nuclide**	**Estuary**	**River**	**Estuary**	**River**
Bone	Ash/wet wt.	0.13	0.10	0.12	0.11
	^{90}Sr	16 ± 10	8 ± 6	12 ± 7	10 ± 6
	Ca	38	35	36	38
	Sr	0.045	0.036	0.042	0.040
Muscle	Ash/wet wt.	0.014	0.014	0.012	0.012
	^{3}H	540 ± 60	3600 ± 100	NA	NA
	^{90}Sr	1.4 ± 0.6	2.1 ± 0.5	NA	NA
	^{106}Ru	15 ± 8	20 ± 10	20 ± 10	18 ± 6
	^{137}Cs	< 25	50 ± 20	19 ± 6	16 ± 6
	Ca	0.35	0.31	0.29	0.33
	Sr	0.0005	0.0004	0.0004	0.0006
	K	3.9	3.9	3.5	3.4
Kidney + liver	^{55}Fe	4700 ± 200	4500 ± 200	NA	NA
	Fe	0.14	0.14	NA	NA
Gut	^{137}Cs	< 30	< 25	< 20	70 ± 30

Notes:

1. ± values are 2-sigma counting error; < values are 3-sigma counting error.
2. NA – not analyzed
3. The following were not detected in muscle, kidney + liver, and gut; (< 30 pCi/kg) ^{58}Co, ^{60}Co, ^{65}Zn, and ^{134}Cs; (< 100 pCi/kg) ^{131}I.
4. kg values are wet weight.

lower potassium content than sea water.(14) The concentration of ^{3}H measured in the shad caught upriver is similar to that measured in fish from the discharge canal, despite the short period of time the shad normally remain in the river just below the station (see Section 5.5.1). Additional sampling is needed to reach any conclusions regarding the levels of these radionuclides in shad.

5.5.4 *Estimated average radionuclide concentrations in fish.* The following radionuclide concentrations and percents of intake limits (discussed in Section 5.5.5) for fish from the coolant canal were computed from average radionuclide concentrations in canal water to indicate the magnitude of and major contributors to the radiation dose from eating these fish:

Radio-nuclide	Annual average concentration in canal water,* pCi/l	Concentration factor(14)	Hypothetical concentration in canal fish,† pCi/kg	Percent of limit**
^{3}H	9,700	0.90	8,700	0.01 TB
^{14}C	0.04	4,500	200	<0.001 TB
^{24}Na	0.03	20	0.6	<0.001 GI
^{32}P	0.01	100,000	1,000	2 B
^{51}Cr	0.007	40	0.3	<0.001 GI
^{54}Mn	0.42	100	42	0.004 GI
^{55}Fe	0.6	100	60	<0.001 S
^{59}Fe	0.01	100	1	<0.001 GI
^{57}Co	0.008	20	0.2	<0.001 GI
^{58}Co	3.6	20	72	0.008 GI
^{60}Co	0.56	20	11	0.002 GI
^{90}Sr	0.0007	5	0.004	<0.001 B
^{95}Zr	0.001	3.3	0.003	<0.001 GI
^{95}Nb	0.004	30,000	120	0.02 GI
^{99}Mo	0.04	10	0.4	<0.001 GI
^{110m}Ag	0.004	2.3	0.009	<0.001 GI
^{131}I	2.0	15	30	1.0 T
^{133}I	0.94	15	14	0.13 T
^{135}I	1	15	15	0.06 T
^{134}Cs	0.1	400	40	0.02 TB
^{136}Cs	0.03	400	12	<0.001 TB
^{137}Cs	0.50	400	200	0.04 TB

* From Section 5.3.2, using values reported by the station operator when available.

† The product of the values in columns 2 and 3. The concentration in river fish would be 1/25th of these values, due to dilution.

**The limit is based on an intake of 50 g fish per day[illegible] that will result in an exposure equal to the Radiation Protection Guides recommended by the FRC[illegible]: The RPG are 500 mrem/yr for thyroid (radioiodine) and 170 mrem/yr for all other critical organs. The critical organs are: (TB) total body; (GI) gastrointestinal tract; (B) bone; (S) spleen; (T) thyroid.

The only radionuclide detected in samples of shad stomach, intestine, thyroid and roe was naturally occurring ^{40}K. In the kidney plus liver, equal amounts of the fallout radionuclide ^{55}Fe (33 ± 3 pCi/mg iron) were observed in river and control fish. The average ^{90}Sr concentration in shad bone of 12 ± 4 pCi/kg was lower than in fresh water fish. All other radionuclides were below the limits of detection given in the note to Table 5.8.

These radionuclide concentrations were computed with the listed concentration factors for edible portions of fresh water fish(14) from the estimated annual average concentrations of radionuclides in the discharge canal water given in Section 5.3.2. Concentrations in water were based on the station's effluent data (first data column of the table in Section 5.3.2) when available; when not, data from this study were used. For the purpose of the calculations, it was assumed that radionuclides in the edible portions of all consumed fish had reached equilibrium with radionuclide concentrations in the discharge canal water. In reality, these calculated radionuclide concentrations in water and concentration factors from water to fish flesh are only approximate, and radioactive equilibrium may not have been attained in the fish.

The computed concentrations in fish agree with the average measured values in canal fish minus those in background fish (see Table 5.7) for ^{14}C, ^{134}Cs, and ^{137}Cs, but are several-fold higher for ^{3}H. The computed concentrations of ^{55}Fe, ^{60}Co, and ^{131}I were below the limits of detection. The computed values for ^{58}Co and ^{95}Nb appear to be too high by at least a factor of two in that the indicated concentration would have been detected if present.

5.5.5 *Estimated population radiation dose.* Phosphorus-32 and the three radioisotopes of iodine are the critical radionuclides according to the table in Section 5.5.4, being at the highest percent of the limit in fish caught in the coolant canal. The annual doses from the listed radionuclides would be 3 mrem/yr to bone (mostly from ^{32}P), 6 mrem/yr to the thyroid (mostly from ^{131}I), 0.4 mrem/yr to the GI tract (mostly from ^{32}P) and 0.3 mrem/yr to the total body. The dose rates from eating 18 kg fish per year estimated in the Environmental Statement are higher for all organs except the thyroid:(33)

Total body	**10.0 mrem/yr**
GI tract	**0.66**
Thyroid	**2.8**
Bone	**7.5**

A total-body dose of 0.7 mrem/yr, almost entirely from ^{134}Cs and ^{137}Cs, is derived from the dose calculations for a model 2-reactor 2,000 MWe PWR station,(32) adjusted 4-fold downward for the lower power level of the Haddam Neck station. This value, however, is based on the different concentration factors in an earlier version of the report(14) used here.

The percentages of intake limits in Section 5.5.4 (last column in the table) are based on the indicated radionuclide concentrations in water. Daily consumption of 50 g of fish caught 6 months per year in the canal (concentrations in column 4 x 0.5 yr x 0.05 kg/day) and 6 months per year in the river (concentrations in column 4 x 0.5 yr x 0.05 kg/day x 0.04 dilution) is assumed. The calculations took the maximum permissible daily occupational drinking-water intake listed by the ICRP to correspond to 5 rem/yr to the total body, 15 rem/yr to the GI tract, and 30 rem/yr to bone,(33) or directly applied FRC guidance for radiostrontium and radioiodine.(30) For these calculated dose rates, the limit is taken to be the Radiation Protection Guides for a "suitable sample" of the exposed population as recommended by the FRC: 500 mrem/yr for the thyroid and 170 mrem/yr for all other critical organs.(30) The applied ratios(34) of pCi/day per rem/yr are tabulated in Appendix E.2.

Of the critical radionuclides, ^{131}I in fish muscle at the indicated concentrations may be barely detectable by gamma-ray spectrometry of fresh fish, but was not measured because of the time that elapsed between sampling and analyses. Measurements of ^{32}P by beta-particle detection after chemical separation were not undertaken. Both of these radionuclides should be measured in future studies.

Annual doses based on measured radionuclide concentrations in fish are tabulated below. The average amounts from Table 5.7 of ^{3}H, ^{14}C, ^{134}Cs and ^{137}Cs measured in canal fish muscle, and of ^{90}Sr in muscle inferred from fish bone analyses, are given for a 50-g sample. The amount in the canal fish due to the station was obtained by subtracting the concentration values in control fish. The annual radiation doses in the next column are based on the ratios in Appendix E.2 for a daily intake of 50 g fish caught 6 months of the year from the canal and 6 months of the year from the river.

Radio-nuclide	**Average concentration measured in fish, pCi/50 g**		**Radiation dose from station, mrem/yr**	**Critical organ**
	Total	**From station**		
^{3}H	145	120	0.005	Total body
^{14}C	45	12	0.002	Total body
^{90}Sr	0.3	< 0.1	< 0.05	bone
^{134}Cs	3	3	0.04	Total body
^{137}Cs	10	6	0.04	Total body

When fallout is included, the dose is 0.13 mrem/yr to the total body, and 0.25 mrem/yr to bone. The additional radiation dose from eating shad that contains 160 pCi ^{3}H per 50 g during 3 months would be 0.003 mrem.

5.6 Radionuclides in Shellfish

5.6.1 *Collection and analysis.* Shellfish were reported to have been in the Connecticut River near the station, but could not be found.(10) Shellfish were sampled from the mouth of the Connecticut River, however, because they are harvested from this location for human consumption. Clams *(Ellito complanatus)* and oysters *(Crassostrea virginica)* were collected on April 16, 1971, from the mouth of the Connecticut River. At the same time, control samples were collected from brackish water near Elihu Island, Stonington, Connecticut, approximately 38 km east of the mouth of the Connecticut River.*

The shellfish were returned to the laboratory on dry ice. The meat was thawed, removed from the shell, and analyzed for photon-emitting radionuclides and radiostrontium as described in Section 5.5.2. The shells were analyzed similarly after removing all organic and chitinous material from their exterior.

5.6.2 *Results and discussion.* No radionuclides attributed to the nuclear power station were observed in any shellfish samples, as shown in Table 5.9. Only ^{40}K, ^{55}Fe, ^{90}Sr and ^{137}Cs were detected, and these were approximately at the same concentrations in samples from the Connecticut River estuary and from Stonington.

Cesium-137 was below the detectable concentration in oyster meat, but was measured in clam meat. This agrees with a report by Chipman who observed in most cases higher ^{137}Cs levels in clams than in oysters.(38) For ^{55}Fe, the concentration was higher in oyster meat: the specific activity was 4.5 ± 0.3 nCi/mg iron. The ^{90}Sr levels were similar in clam and oyster meat. The average concentration was 70 ± 15 pCi/kg, about 10 times the concentration in the fish muscle.

The only radionuclides in the shells were ^{55}Fe, at a specific activity of 6.4 ± 1.9 pCi/mg Fe in the clam, and ^{90}Sr in oyster and clam. The concentration of ^{55}Fe was approximately 20 times lower than in meat, but the ^{90}Sr concentration was 10 times higher.

The average concentrations of ^{55}Fe, ^{90}Sr and ^{137}Cs measured in shellfish muscle correspond to the following annual radiation exposure at a daily consumption of 50 g:(36)

Table 5.9

Radionuclide (pCi/kg) and Stable Ion (g/kg) Concentrations in Shellfish

	Oysters		Clams	
Nuclide	**River mouth**	**Control**	**River mouth**	**Control**
Meat				
^{55}Fe	127,000 ± 38,000	200,000 ± 36,000	30,000 ± 20,000	50,000 ± 25,000
^{90}Sr	80 ± 20	80 ± 15	60 ± 15	50 ± 20
^{137}Cs	< 25	< 30	60 ± 20	60 ± 30
Ca	1.6	0.48	8.2	0.93
Sr	0.029	0.010	0.090	0.013
Fe	0.029	0.045	0.025	0.051
K	0.8	0.7	1.1	0.6
Shell				
^{55}Fe	< 300	< 330	1500 ± 400	2000 ± 500
^{90}Sr	640 ± 60	620 ± 60	560 ± 50	980 ± 70
Ca	150	170	400	440
Sr	0.34	0.50	0.70	1.9
Fe	0.09	0.12	0.30	0.26

Notes:

1. ± values are 2-sigma counting error; < values are 3-sigma counting error.
2. The following were not detected:
   in meat—^{60}Co (< 40 pCi/kg), ^{131}I (< 110 pCi/kg), ^{134}Cs (< 40 pCi/kg)
   in shell—^{60}Co (< 40 pCi/kg), ^{65}Zn (< 70 pCi/kg).
3. kg values are wet weight.

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*We thank R. Massengill for collecting the shellfish samples.

Radionuclide	Average concentration in shellfish, pCi/50 g	Radiation dose, mrem/yr	Critical organ
^{55}Fe	5,100*	4	Spleen
^{90}Sr	3.5*	3	Bone
^{137}Cs	3**	0.03	Total body

*Average concentration in clam and oyster meat.
**Average concentration in clam meat only.

According to this tabulation, a person who consumes 50 g shellfish per day will receive a dose to the spleen and bone of 4 and 3 mrem/yr, respectively. Because no significant differences were observed between the radionuclide concentrations in shellfish collected from the mouth of the Connecticut River and in the control samples, these exposures are attributed to fallout and not to the Haddam Neck station.

On the assumption that some clams can be obtained for consumption from the discharge canal, the Environmental Statement attributed to station operation doses of 0.1–0.2 mrem/yr to the total body, thyroid, GI tract, and bone.(31)

5.7 Radionuclides in Sediment

5.7.1 *Sampling and measurement.* Bottom sediments in the coolant water discharge canal and near the banks of the Connecticut River between Haddam Island and the Salmon River were surveyed with a submersible gamma-ray probe on September 14–16, 1970. The 10-cm x 10-cm-dia. cylindrical NaI(Tl) detector in a water-tight container was connected to a multichannel analyzer operated at 10 keV/channel.* The analyzer and associated equipment, including portable electric generator, were operated in a boat. The probe was lowered over the side of the boat and then placed by a diver. Most of the locations at or below the station (see Figure 5.3) were selected because staff from the Essex Marine Laboratory had observed that suspended material accumulated there. Three locations near Haddam Island were selected as controls, to measure the natural background radiation in the vicinity.

Probe readings were screened initially by observing dead-time values; those that were above 1 percent suggested the presence of radiation levels higher than usual, possibly from effluents at the station. A gamma-ray spectrum was obtained for a 10-min period at each of these locations and at the background sites. The background spectra showed gamma rays from ^{40}K and the progeny of radium and thorium. Gamma-rays attributed to ^{58}Co and ^{60}Co discharged at the station were observed in spectra that showed the higher dead times.

At the locations indicated by the letter B in Figure 5.3, the diver collected sediment samples. Two samples were generally taken side by side—the top 3 cm scraped with a specially devised scoop, and a 10-cm-dia. core, 13 cm deep. These samples were identified by the letters T and C, respectively. The wet samples were stored in plastic containers for analysis.

Two additional sediment samples were collected on March 2, 1971, from the mouth of the discharge canal (B-31) and from the mouth of the discharge canal at the Maromas fossil-fueled plant near Middletown, 18 km upstream (B-30). These two samples were obtained with a Petersen dredge. A core sample (B-32), 86 cm deep and 4.5 cm in diameter, was collected on April 28, 1971, from the accumulated sediment at the mouth of the canal. The sampling location was intended to be that of sample B-7. This core sample contained five separate zones and was later sectioned into these five parts.

All samples were returned to the laboratory, air dried at room temperature (20° C), homogenized and passed through a 2.0-mm (No. 10) sieve. Material larger than 2 mm, mostly twigs and pebbles, was discarded.

5.7.2 *Description of sediment samples.* To define geochemically the nature of the collected sediment samples, some were analyzed for pH, organic content, particle size distribution, cation exchange capacity, and clay mineralogy.† The results of these analyzes are presented in Table 5.10.

The analytical techniques are standard procedures of ASTM and the American Society of Agronomy.(37) In brief, the pH was determined by stirring a weighed aliquot of dried, homogenized sample with an equal volume of water, and reading the pH of the supernatant

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*We thank Mr. Charles Phillips, Eastern Environmental Radiation Facility, EPA, for providing the instrument and participating in the measurements

†We thank Professor L. Wilding, Department of Agronomy, Ohio State University, for performing these analyses.

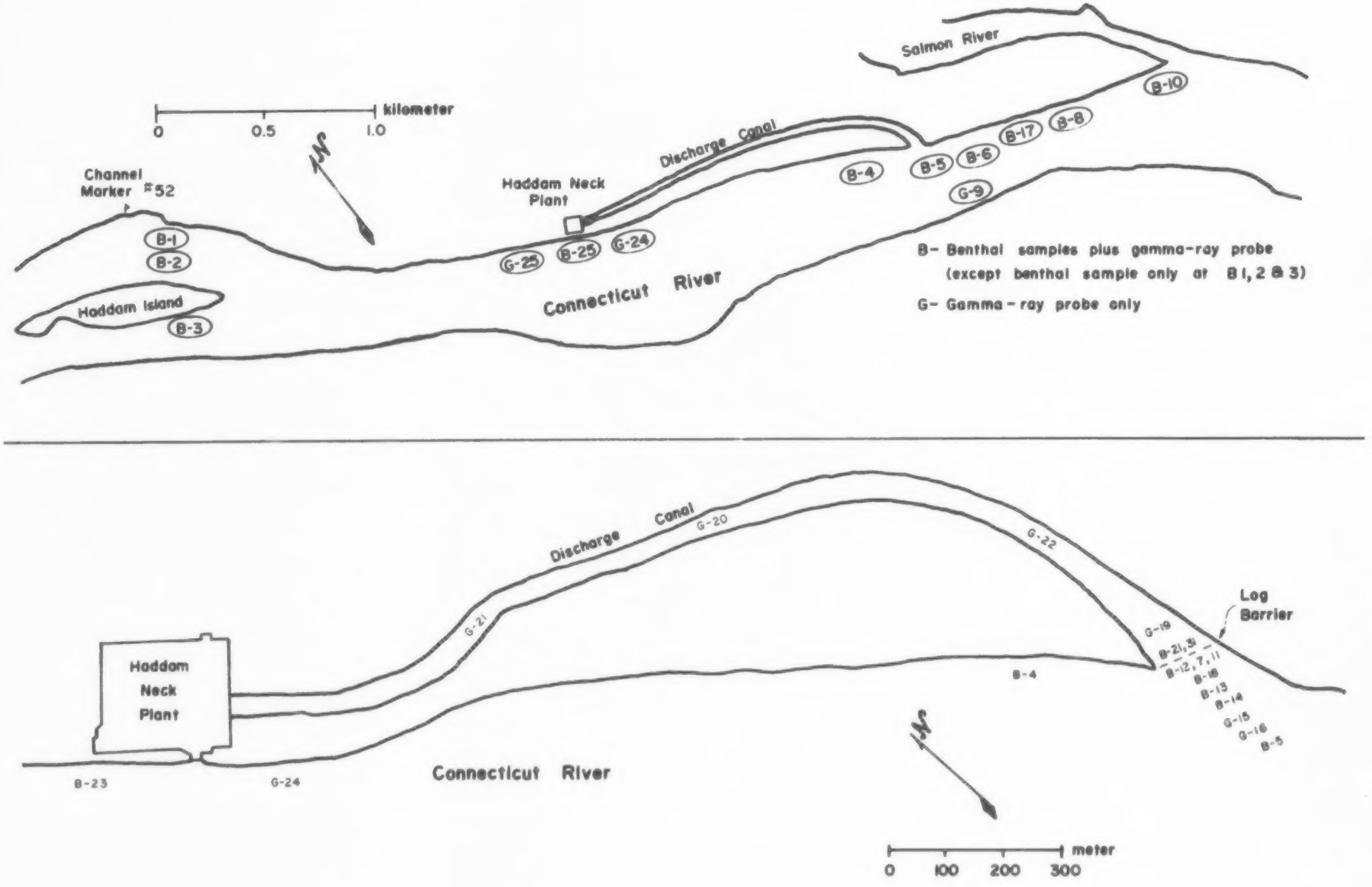
Salmon River
B-10
B-8
B-17
B-6
B-5
G-9
B-4
Discharge Canal
0
0.5
1.0
kilometer
N
Channel Marker #52
Haddam Neck Plant
B-1
B-2
Haddam Island
B-3
G-25
B-25
G-24
Connecticut River
B- Benthal samples plus gamma-ray probe (except benthal sample only at B 1, 2 & 3)
G- Gamma-ray probe only
Discharge Canal
G-20
G-22
G-21
Log Barrier
G-19
B-21, 31
B-12, 7, 11
B-18
B-13
B-14
G-15
G-16
B-5
B-4
Haddam Neck Plant
B-23
G-24
Connecticut River
N
0
100
200
300
meter

Figure 5.3 Sites for Sediment Sampling and Gamma-Ray Probe Measurements

Table 5.10

Mineralogical Analyses of Sediment Samples

Sample No.	2	5	12	12	17	21	31
Category	Top	Top	Core	Top	Core	Core	Dredge
Texture	sand	sand-to-loamy sand	loamy sand	loamy sand	sandy loam	sand	loamy sand
pH	5.8	5.3	5.4	5.4	5.8	6.0	5.6
	Organic Content, % of Total Dry Weight						
Organic carbon	0.33	1.06	0.60	0.68	2.50	0.55	0.79
Organic matter	0.57	1.82	1.03	1.17	4.30	0.95	1.36
	Particle Size Distribution, % of Total Weight						
Clay (< 2 μ dia.)	<0.1	4.3	1.4	1.4	2.0	1.6	4.0
Silt (2–50 μ dia.)	1.8	8.4	18.4	16.2	44.6	4.3	13.0
Sand (50–2000 μ dia.)	98.2	87.3	80.2	82.4	53.4	94.1	83.0
	Cation Exchange Capacity (CEC), meq/100 g						
Total, Direct Method	2.2	2.3	3.6	4.2	8.5	2.2	1.8
Exchangeable H^+	1.4	---	4.0	3.2	8.8	3.6	--
Exchangeable Bases	0.6	---	1.1	1.4	5.0	1.4	--
Sum exchange H^+ + Bases	2.0	---	5.1	4.6	13.8	5.0	--
	Clay Minerals and Amorphous Material, % of Crystalline Fraction*						
Illite (mica)	---	49(37)	---	---	60(44)	53(42)	--
Vermiculite	---	18(13)	---	---	17(12)	19(15)	--
Quartz	---	26(20)	---	---	13(9)	22(17)	--
Chlorite	---	< 5(< 4)	---	---	< 5(< 4)	< 5(< 4)	--
Kaolinite	---	7(5)	---	---	10(7)	6(5)	--
Amorphous Material**	---	25	---	---	28	21	--

*values in parentheses are % of crystalline fraction, that is % of total clay fraction minus organic matter and allophane contents.

**% of non-organic matter.

liquid with a standard glass electrode. The carbon content of the samples was determined by a gravimetric measurement of the CO_2 produced from thermal decomposition of the dried material. The carbon values were converted to equivalent organic matter by multiplying by 1.72. Particle-size distribution was determined for the sandy fractions (> 50 micron) by dry sieving, while the finer silt and clay fractions (< 50 micron) were digested with H_2O_2 and the Calgon-dispersed aliquots were separated by sedimentation. The total cation exchange capacity (CEC) was measured directly by saturating a weighed sample with K^+ (as KCl), eluting the excess salt with 95-percent ethanol, displacing the K^+ ion with NH_4^+ (NH_4OH at pH 7), and measuring the K^+ ion by an atomic absorption spectrophotometer. To confirm these values, the CEC of five samples was also measured indirectly as the sum of exchangeable bases (saturating with sodium acetate) and exchangeable acids (barium chloride-triethanolamine method). The exchangeable acid is H^+, while the exchangeable bases are defined as the alkali and alkaline earth metals.

Sample B-17 was unusual in that the alcohol wash was highly colored, suggesting removal of some organic matter that could account for a fraction of the total CEC. For this sample, the CEC obtained by the sum technique was considered more valid, although, even for the other samples, values obtained by the different techniques are not necessarily comparable.

The clay mineral composition of three samples was determined by X-ray crystallography of preferred-oriented aggregate clay fractions on ceramic plates.(37) The sample particles were dispersed with Na_2CO_3 without prior removal of carbonates and iron oxide, and the <2-micron clay fractions were flocculated with NaCl. Amorphous matter was taken as the difference between total sample weight and the sum of the crystalline clay fractions.

It was not possible, on the basis of these data alone, to distinguish CEC due to organic matter from that arising from the mineral component. However, data in Tables 5.10 and 5.11 for sample #31 suggests that most of the exchange capacity was associated with the clay fraction.

To determine the effects of sample preparation on particle size distribution, percent of clay minerals and organic matter, and total cation exchange capacity, sample B-31 was analyzed in the following three forms; (1) air-dried and electrolyte dispersed, (2) air-dried and water dispersed, (3) wet and electrolyte dispersed. The results, given in Table 5.11, indicate that evaluations of the clay fraction are dependent on sample preparation. Hence, carefully defined standardized analytical techniques must be used in studies of this type.

Table 5.11

Comparative Analyses of Sediment Sample B-31

	Air Dried		Wet
Analysis	**Electrolyte dispersed**	**H_2O dispersed**	**Electrolyte dispersed**
Particle Size Distribution			
Clay (< 2μ dia.), %	4.0	2.8	3.8
Silt (2-50μ dia.), %	13.0	16.4	14.4
Sand (5—2000μ dia.), %	83.0	80.7	81.8
CEC, meq/100 g	1.8	---	2.7
CEC, clay fraction only, meq/100 g	59	32	--
Organic carbon, clay fraction, %	7.0	5.9	--
Organic matter*, %	12.6	10.6	--

*% carbon x 1.72

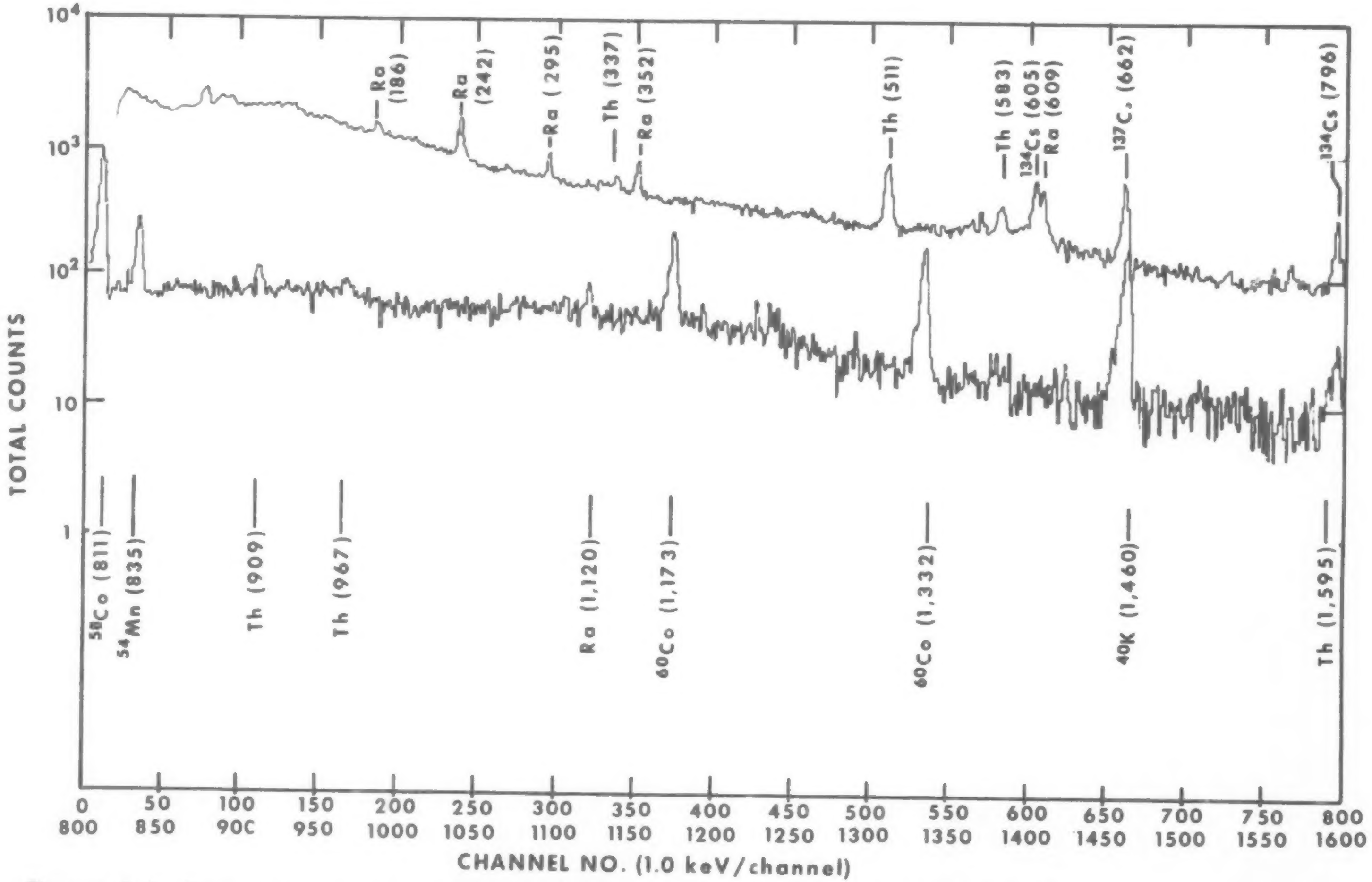

Figure 5.4 Gamma-ray spectrum of sediment sample, B-7 (top), 0-1600 keV.

Detector: Ge(Li), 10.4 cm^2 × 11 mm, trapezoidal.
Sample: 564 gm (400 cc) dry wt., collected Sept. 13, 1970 from mouth of discharge canal.
Count: Nov. 9-10, 1970 (1000 min., background not subtracted); Ra and Th refer to ^{226}Ra and ^{232}Th plus progeny.

5.7.3 *Radioactivity measurement.* The count rates due to ^{58}Co and ^{60}Co in the gamma-ray spectra obtained with the submersible probe were computed by subtracting interferences from other radionuclides. For ^{58}Co, net count rates were derived by summing the energy range from 0.76 MeV to 0.88 MeV, and subtracting the Compton interference estimated graphically under each ^{58}Co peak by tracing a straight line over the energy interval from 0.50 to 1.00 MeV. Interference from traces of ^{54}Mn (0.84 MeV) and ^{134}Cs (0.80 MeV) was not considered. This interference was determined later from laboratory analyses of sediment collected at the probe points to have resulted in ^{58}Co values overestimated by no more than 25 percent. Net count rates due to ^{60}Co were calculated for its 1.17-MeV and 1.33-MeV photopeaks together by summing from 1.08 MeV to 1.37 MeV and subtracting as background over the same energy range the average count rates for probe measurements at locations 1, 2, 3, 5, and 6. The major correction is for the 1.46-MeV gamma ray of naturally occuring ^{40}K.

Radionuclides in the sediment samples were measured by analyzing 400 ml volumes(360–680 g) of air-dried, homogenized sample by gamma-ray spectrometry with a 10– x 10-cm NaI(Tl) detector or a Ge(Li) detector (see Figure 5.4). The NaI(Tl) spectra were solved by a computer (matrix technique) program. The better-resolved Ge(Li) spectra were solved by direct calculation. Although sample densities ranged from 1.3 to 1.7 g/cm^3, no correction for self-absorption was applied because the effect of absorption in this density range on measuring the radiations of interest was found to be no larger than 10 percent.

Concentrations of ^{90}Sr were determined by separating the strontium from either acid-leached or fused samples with an ion-exchange procedure.(19) The ^{90}Sr was determined by counting the beta particles of the ingrown ^{90}Y.

The initial analyses of the sediment samples for naturally occurring ^{226}Ra were with the NaI(Tl) detector, based on the 1.76-MeV photon of the ^{214}Bi progeny. Radioactive equilibrium between parent and daughter was assumed. Later analyses of most of these samples with the Ge(Li) detector by measuring directly the 0.186-MeV photon of ^{226}Ra indicated that earlier results yielded only about 40 percent of the actual ^{226}Ra concentration. This presumably was due to the emanation of ^{222}Rn from the dried samples. Samples 2, 3, 5, and 6 were only measured by NaI(Tl) detector and, therefore, have ^{226}Ra results that are too low to this extent.

5.7.4 *Results and discussion of analyses.* Sediment collected at the mouth of the discharge canal and at selected locations immediately upstream and downstream along the eastern shore of the Connecticut River (see Figure 5.3) contained radionuclides attributable to plant operation as shown in Table 5.12. The highest levels were associated with particulate sediment deposited in the mouth of the discharge canal. The four background samples (1, 2, 3 and 30) collected in the river upstream from the station contained ^{54}Mn, ^{90}Sr, and ^{137}Cs—presumably from fallout—at concentrations not exceeding 0.2 pCi/g. They also contained naturally occurring ^{40}K, ^{226}Ra plus progeny and ^{232}Th plus progeny. No ^{57}Co, ^{58}Co, ^{60}Co, or ^{134}Cs ($<$0.1 pCi/g) was detected in these 4 samples and in sample 6.

Cobalt-58 was the predominant radionuclide indicative of contamination from plant discharges in the sediment samples. It was usually accompanied by ^{60}Co, ^{134}Cs, and ^{137}Cs. Some of the samples also contained ^{54}Mn and ^{57}Co. Only samples 5 and 31 contained some of these radionuclides without ^{58}Co. Concentrations of ^{90}Sr above background were not found, except possibly in sample 17C.

The ratio in Table 5.13 of radionuclide concentrations in the top 3 cm to those in the top 13 cm of sediment from the mouth of the discharge canal indicate approximately 2-fold higher levels of effluent radionuclides near the top. For the naturally occurring radionuclides ^{40}K, ^{226}Ra and ^{232}Th, the ratio was near unity. Concentrations of effluent radionuclides in the sections of core sample 32 (Table 5.14) also were highest near the surface. None of these radionuclides were found below the 29-cm depth.

5.7.5 *Results and discussion of probe measurements.* The probe was a convenient device for surveying benthic matter *in situ* because results were immediately available to indicate the location and distribution of radioactivity. Analysis of sediment collected at the locations examined by the probe then provided precise information concerning radionuclide identity and quantity. An approximate value of the counting efficiency of the probe was obtained by comparing the probe count rate with radionuclide concentrations in samples from the same locations.

The probe measurements in the Connecticut River and discharge canal detected ^{58}Co and ^{60}Co in the mouth of the discharge canal and for short distances up- and down-stream along the east shore of the Connecticut River (see Table 5.15). Neither ^{58}Co nor ^{60}Co was detected by the probe within the discharge canal. The ^{58}Co and ^{60}Co count rates were highest at the

Table 5.12

Concentration of Radionuclides in Sediment, pCi/g dry weight

Sample No.	Sample Wt., g/400 cc	^{54}Mn	^{58}Co	^{60}Co	^{134}Cs	^{137}Cs	^{90}Sr	^{40}K	^{226}Ra	^{232}Th
1C	621	<0.1	<0.1	<0.1	<0.1	0.2	0.2	10.8	0.2	0.3
1T	603	<0.1	<0.1	<0.1	<0.1	0.1	0.1	10.8	0.3	0.3
2C	595	0.2	<0.1	<0.1	<0.1	<0.1	0.1	10.5	0.3	0.3
2T	662	0.2	<0.1	<0.1	<0.1	0.2	0.2	9.0	0.7	0.5
3C	622	0.1	<0.1	<0.1	<0.1	0.2	0.1	10.8	0.2	0.2
3T	572	<0.1	<0.1	<0.1	<0.1	0.1	0.1	11.0	0.3	0.3
4C	633	<0.3	0.4	<0.1	<0.1	<0.1	<0.1	9.8	1.0	0.3
4T	608	<0.2	0.6	<0.1	<0.1	<0.1	0.1	11.0	0.6	0.3
5C	500	<0.1	<0.1	<0.1	0.1	0.5	0.1	9.9	0.5	0.5
5T	479	<0.1	<0.1	<0.1	<0.1	0.6	0.1	9.9	0.5	0.5
6C	623	<0.1	<0.1	<0.1	<0.1	0.1	0.1	9.7	0.2	0.3
6T	645	<0.1	<0.1	<0.1	<0.1	0.1	0.1	10.1	0.2	0.3
7C	592	0.5	4.1	0.9	0.6	1.3	0.1	10.1	1.1	0.4
7T	564	<0.2	5.6	1.1	0.6	1.4	0.1	9.5	1.5	0.4
8C	481	<0.3	1.3	0.2	<0.1	0.6	0.2	10.0	1.5	0.6
8T	459	<0.3	3.3	0.4	0.3	0.8	0.2	9.2	1.3	0.6
10C	428	<0.4	0.5	0.2	<0.1	0.6	NA	14.3	1.8	0.7
10T	444	<0.3	2.5	<0.1	<0.1	0.6	NA	11.3	2.3	0.6
11C	590	0.3	2.8	0.6	0.5	0.7	0.1	11.7	1.0	0.3
11T	629	<0.2	1.4	0.3	0.1	0.2	0.1	8.6	0.5	0.3
12C	545	<0.1	1.9	0.5	0.4	0.8	NA	12.5	1.5	0.3
12T	546	1.5	13.5	3.2	1.9	3.1	0.1	10.5	1.0	0.5
13C	443	0.1	8.1	1.6	1.1	1.9	0.2	13.5	1.0	0.5
14C	483	<0.3	2.5	0.6	<0.1	0.7	0.1	10.6	1.2	0.5
17C	364	1.4	8.5	1.5	1.0	2.0	0.3	11.7	1.0	0.8
18C	562	0.5	4.3	0.8	0.4	0.7	0.2	9.2	0.7	0.5
21C	585	0.7	5.5	1.2	0.6	1.3	0.1	9.5	1.5	0.3
23C	447	<0.1	1.4	0.2	0.4	0.9	NA	11.1	0.7	0.7
30D	515	<0.1	<0.1	<0.1	<0.1	0.2	0.1	14.5	0.7	0.3
31D	484	<0.2	<0.1	0.1	0.3	0.4	0.1	13.1	1.5	0.4

Notes:

1. Analyses of gamma-ray emitters are by Na I(Tl) spectroscopy for samples 2, 3, 5, and 6 and by Ge(Li) for all other samples [except that all ^{232}Th and ^{58}Co analyses are based on the initial NaI(Tl) scan, corrected for long-lived interference with ^{60}Co by a later Ge(Li) scan].
2. T = top 3 cm (0–3 cm depth); C = top 13 cm (0–13 cm depth); D = by Petersen dredge.
3. Detection limits at 2-sigma counting error were ± 0.1 pCi/g except: ^{54}Mn by Ge(Li) = ± 0.2 pCi/g, ^{40}K by NaI(Tl) = ± 0.3 pCi/g, ^{40}K by Ge(Li) = ± 1.2 pCi/g, ^{226}Ra by Ge(Li) = ± 0.4 pCi/g.
4. ^{57}Co was found at a concentration of 0.04 pCi/g in sample 7T and 0.09 pCi/g in 12T.

Table 5.13

Concentration of Radionuclides in "Core" vs. "Top" Samples*

Radionuclide	Average concentration, pCi/g 13-cm depth (C)	Average concentration, pCi/g 3-cm depth (T)	Ratio, T/C
^{58}Co	2.7	5.9	2.2
^{60}Co	0.44	1.0	2.3
^{134}Cs	0.28	0.46	1.6
^{137}Cs	0.60	1.1	1.8
^{40}K	11.7	9.8	0.8
^{226}Ra	1.4	1.3	0.9
^{232}Th	0.50	0.48	1.0

*Samples B-7, B-8, B-10, B-11, and B-12 from mouth of discharge canal (see Figure 5.2).

Table 5.14

Concentration of Radionuclides in Core Sample (B-32) as Function of Depth, pCi/g dry weight

Depth, cm	Weight, g	^{58}Co	^{60}Co	^{134}Cs	^{137}Cs	^{40}K	^{226}Ra
0–6	55.4	0.3	0.38	0.4	1.1	12.4	0.3
6–18	113.6	<0.1	0.06	0.10	0.26	11.7	1.5
18–29	120.0	<0.1	0.15	0.09	0.24	9.5	1.1
29–44	153.9	<0.1	<0.02	<0.03	<0.02	9.7	1.0
71–79	122.0	<0.1	<0.03	<0.03	<0.02	10.0	0.9

Notes:

1. ^{54}Mn was not detected (<0.1 pCi/g).
2. 2-sigma ± values are 0.04 pCi/g for ^{60}Co, ^{134}Cs, and ^{137}Cs, 0.1 pCi/g for ^{58}Co, and 0.4 pCi/g for ^{40}K and ^{226}Ra; <values are 3-sigma of counting error.
3. No ^{58}Co, ^{60}Co, ^{134}Cs, and ^{137}Cs were detected in samples from depths between 44 and 70 cm.
4. Values at 0–6 cm were obtained from a NaI(Tl) spectra; all others, from Ge(Li) spectra.
5. Sample depths correspond to visibly separate zones.

Table 5.15

Net Count Rate of ^{58}Co and ^{60}Co with NaI(Tl) Underwater Probe

Location	^{58}Co, Counts/min	^{60}Co, Counts/min	Location	^{58}Co, Counts/min	^{60}Co, Counts/min
B-1	<20	<20	B-14	550 ± 30	430 ± 20
B-2	<20	<20	G-15	1020 ± 40	680 ± 20
B-3	<20	<20	G-16	570 ± 30	420 ± 20
B-4	20 ± 10	<20	B-17	500 ± 30	570 ± 20
B-5	<20	<20	B-18	1100 ± 40	670 ± 20
B-6	<20	<20	G-19	880 ± 40	610 ± 20
B-7	1110 ± 40	690 ± 30	G-20	<20	<20
B-8	140 ± 20	80 ± 20	G-21	note 4	note 4
G-9	<20	<20	B-21	770 ± 30	310 ± 20
B-10	<20	<20	G-22	note 4	note 4
B-11	410 ± 30	290 ± 20	G-24	<20	<20
B-12	690 ± 30	560 ± 20	G-25	<20	<20
B-13	920 ± 40	640 ± 20			

Notes:

1. See Figure 5.3 for locations.
2. All counting times were 10 min.
3. ± values are 2-sigma counting error; < values are 3-sigma counting error.
4. Gamma-ray spectra were not recorded because counter dead-time indicated zero.

mouth of the discharge canal.

The counting efficiencies of the probe for ^{58}Co and ^{60}Co, given in Table 5.16 in terms of the ratio of probe count rate of Table 5.15 to the concentrations of Table 5.12, varied considerably among locations. This would be expected from nonuniform vertical and horizontal distributions of radionuclides in the sediment. The response of the probe did not differ significantly with respect to the radionuclide content in samples from a depth of 0–3 cm ("top") or 0–13 cm ("core"). The average ratios for ^{60}Co of 500 and 600 c/min per pCi/g are consistent with values between 290 and 800 c/min per pCi/g observed in the reservoir at Yankee-Rowe.(5)

The sensitivity of the probe is 0.15 pCi/g for ^{58}Co and 0.04 pCi/g for ^{60}Co, based on the minimum detectable count rate above background of 20 c/min (see Table 5.15) and the respective average ratios in Table 5.16 of 130 and 550 c/min per pCi/g. Minimum detectable values based on comparing the relatively low ^{58}Co and ^{60}Co concentrations in samples 4, 8, and 10 with net count rates by the probe are approximately three times as large, and probably more realistic.

5.7.6 *Significance of radioactivity in sediments.* The observation of radioactivity at only a few locations, mainly in the mouth of the discharge canal, suggests that fine particles with associated radionuclides are swept down the discharge canal at a rate too high for deposition. Upon reaching the broad mouth of the canal, the flow rate is slowed and the fine particles are deposited. The Essex Marine Laboratory has observed the growth of the deposit since the station began operating.(17) Radionuclides from the station were also found in lower amounts at places where silt accumulated near the east bank of the Connecticut River extending from the station's water intake upstream from the canal to the mouth of the Salmon River, downstream.

The concentrations of radionuclides in the largest deposit, at the mouth of the canal, were as follows, based on the averages of values measured in core samples 7, 11, 12, 13, 14, 17, 18, and 21 (Table 5.12):

Radionuclide	Average concentration, pCi/g	pCi/cm^2	Estimated total, mCi
^{54}Mn	0.4	9	0.3
^{58}Co	4.7	100	3.5
^{60}Co	0.8	18	0.6
^{134}Cs	0.6	13	0.5
^{137}Cs	1.2	26	0.9

The concentrations per unit area were computed from the average density of 1.3 g/cm^3 and the estimate that the amount of radionuclide in the entire silt column was approximately 1.3 times the amount in the top 13 cm core sample (see Table 5.13). The totals were estimated for an area of 3.5 x 10^7 cm^2; divers from the Essex Marine Laboratory had observed the deposit to be in the shape of a lens approximately 90 m x 60 m just

Table 5.16

Ratio of Count Rate by Underwater Probe to Radionuclide Concentration in Sediment Samples

Location	^{58}Co,(c/min)/(pCi/g)		^{60}Co,(c/min)/(pCi/g)	
	"core"	"top"	"core"	"top"
B-4	50	30	---	--
B-7	270	120	770	630
B-8	110	20	400	200
B-11	150	210	480	970
B-12	360	40	1120	180
B-13	120		400	
B-14	220		720	
B-17	60		380	
B-18	260		840	
B-21	140		260	
range	50–360	20–210	260–1120	180–970
Average (± 1σ)	170±100	80±80	600±280	500±380

Note: Count rates are from Table 5.15 and radionuclide concentrations from Table 5.12.

outside the canal barrier, with a neck extending downriver near the east bank.

The amounts in the sediment are approximately 0.1 percent of these radionuclides in liquid discharges if it is assumed that the effluents since operation began contained twice the discharges reported for the years 1970 and 1971 in Section 4.1.2. This is unlike the situation at the Yankee-Rowe station,(5) where a large fraction of these effluent radionuclides remained in the sediment.

The radiation exposure from these radionuclides in sediment is believed to be minute as long as the deposit is covered by 0.5 m or more of water to absorb the radiation, persons stay at a distance, and the radionuclides remain in place. In the absence of covering water, direct radiation doses to adults 1 m above the benthal radioactivity are related to the surface concentration by factors of 0.042 to 0.17 μrem/hr per pCi/cm^2 for the listed radionuclides, if an infinite plane source is assumed.(38) The accumulating sediment at the mouth of the canal was covered by 4 m of water at the time of the study, and neither fishermen nor swimmers came within its immediate vicinity.

The possibility of radionuclides in benthal material entering the food chain through uptake by fish has been suggested.(4) The extent and rate of such entry into the food chain in the river are not known. Consequently, the deposits are a potential source of radionuclides to the aquatic environment and should be evaluated periodically.

5.8 References

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6. RADIONUCLIDES IN ENVIRONMENTAL AIR

6.1 Introduction

6.1.1 *Purpose.* Studies were conducted in the environs of the Haddam Neck station to test techniques for measuring ambient concentrations of, and radiations from, radionuclides in airborne emissions from a PWR. Tests were performed during three periods of elevated radioactivity release when stored gas was discharged from the waste surge sphere.

Radionuclides routinely discharged to air at the Haddam Neck station are usually dispersed to concentrations near or below detection limits before reaching the ground. For example, the continuous release from the vent stack of the most abundant radionuclide, ^{133}Xe, at a rate of 53 µCi/s (see Table 3.9) would lead to an average concentration of only 2 x 10^{-4} µCi/m^3 at the nearest exclusion fence. The average dispersion value of 3.5 x 10^{-6} s/m^3 (see Appendix D.1) used for this station is probably too large because the elevation of the point of release is not considered. The concentration value is below the minimum detectable level of 4 x 10^{-4} µCi/m^3 by the procedure described in this section. During releases of stored gas from the surge sphere, however, the rate of ^{133}Xe discharges was approximately 40-fold higher. On those occasions, the radionuclide should be readily detected under usual conditions.

The radioactive gases in the waste surge sphere consist primarily of 10.7-year ^{85}Kr and 5.3-day ^{133}Xe, with relatively small amounts of other radionuclides (see Table 3.2). Because ^{85}Kr emits its 514-keV photon with an abundance of only 0.43 percent, almost the entire gamma-ray flux from this gas is produced by ^{133}Xe. The latter emits 32-keV cesium X$_K$ rays and 81-keV gamma rays with 47 and 37 percent abundances, respectively.(1)

Optimum sampling locations were selected by detecting the radiation from the plume of stack effluent during discharge of the sphere with portable scintillation detectors responsive to low-energy radiation. Ambient ^{85}Kr and ^{133}Xe were then sampled by compressing approximately 0.5 m^3 of air into cylinders for analysis at the laboratory. The direct radiation was measured continuously during the collection periods with the scintillation detector coupled to a chart recorder.

6.1.2 *Environment and meteorology.* The immediate environment is described in Section 7.1.2. The following aspects are pertinent to these measurements: the local region is hilly and heavily forested; the plant is on the east bank of the Connecticut River; a steeply rising wooded hill borders the plant on the northwest side and lies within the exclusion area to the east; and an access road from the northwest leads to the site entrance, 530 m northwest of the reactor. Overall, few roads exist in the area. A contour map of the area is given in Figure 6.1.

A site meteorological study before plant construction began showed that winds are affected locally by the hills and tend to follow the river valley. Surface winds, therefore, are predominantly northwesterly or east-southeasterly. Atmospheric dilution from stack to ground level beyond the plant boundary is calculated by the station operator with the diffusion model(2) described in Appendices D.1 and D.2.

A meteorological tower is located on site approximately 200 m south of the stack. Its base is 6 m above mean sea level (MSL). During this study, wind speed and direction were measured 39 m (129 ft) above MSL, and temperature, 31 m (101 ft) above MSL. The data are recorded continuously in the reactor control room.

The 53-m stack is at the same elevation and stands adjacent to the reactor containment structure. The plume will, therefore, be influenced by mechanical turbulence around the building to an extent dependent on meteorological conditions. Stack effluents are most likely to be drawn into the reactor building wake during southeasterly winds. At equal distances near the stack, ground level concentrations nearby would then be higher toward the northwest.

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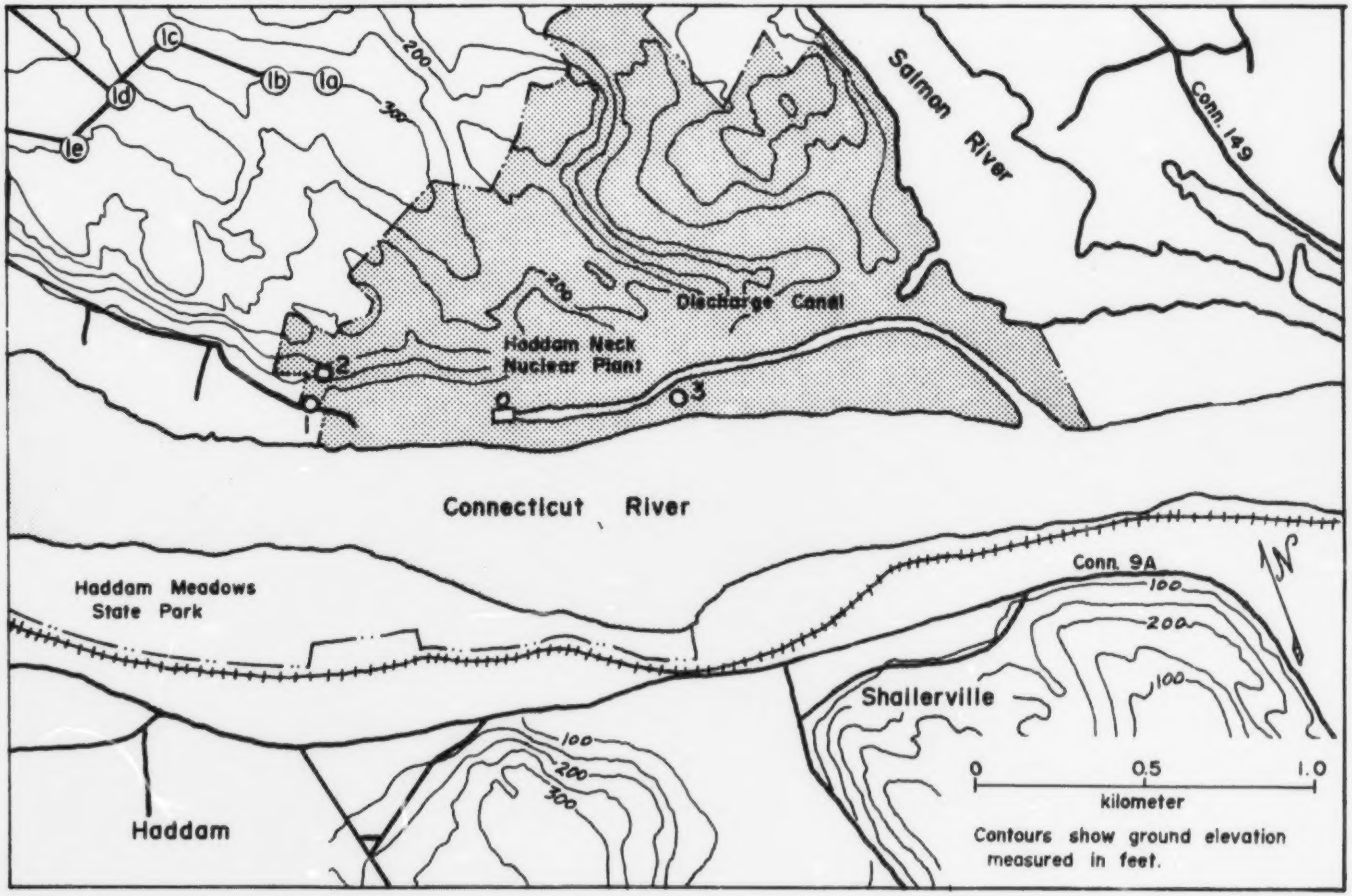
Conn. 149
Salmon River
Discharge Canal
Haddam Neck Nuclear Plant
200
300
1a
1b
1c
1d
1e
1
2
3
Connecticut River
Haddam Meadows State Park
Conn. 9A
100
200
300
Shailerville
Haddam
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0
0.5
1.0
kilometer
Contours show ground elevation measured in feet.

Figure 6.1 Locations for Plume Sampling and Measurement
(Numerals Indicate Test Designation)

6.2 Measurement of Short-term Radiation Exposure and Radionuclide Concentration

6.2.1 *Air sampling.* Samples were obtained with an air compressor (27-V DC Cornelius Model 32-R-300) connected to a 34-liter low-pressure gas bottle rated to contain 0.9 m^3 at maximum pressure. Each cylinder was filled with about 0.5 m^3 air in a period of 40 min. The pump is powered by an 115-V AC motor generator with output converted to 27V DC by a full-wave rectifier.

For ^{133}Xe analysis, sampled air was released from the tank at a rate of 6 liters/min for 16.7 min. It was passed through beds of Linde 13X molecular sieve and Ascarite for removal of water vapor and CO_2, then through a 1-cm-dia. x 80-cm copper cooling coil, and finally through a 3.2-cm-dia. x 66-cm copper U-tube containg 180 g of Columbia 6GC (10–20 mesh) charcoal. Both tubes were immersed in a –76° C dry-ice-acetone refrigerant bath. The charcoal under these conditions collected all ^{133}Xe from 1 m^3 or less of air.

After passage of 100 liters, the U-tube was opened and the charcoal transferred to 10-cm-dia., 450-cc plastic containers. The charcoal was allowed to warm for 1 hour to room temperature to eliminate pressure build-up. The container was then sealed with a rubber gasket and a bolted lid. It was found that 36 percent of the ^{133}Xe on the charcoal is lost due to warming. The charcoal was analyzed for 1000 min with a 10– x 10-cm NaI(Tl) gamma-ray detector connected to a 200-channel spectrometer. The analyzer was calibrated with a ^{133}Xe gas standard provided by the National Bureau of Standards in October 1973.

The remainder of the air sample was analyzed for ^{85}Kr, adding 1.86-hr ^{83m}Kr to determine the krypton yield. Krypton was separated and purified by cryogenic fractionation. The krypton fraction was dissolved in liquid scintillator solution, and ^{85}Kr and ^{83m}Kr were measured in a liquid scintillation counter.*(3)

6.2.2 *Scintillation detector for low-energy photons.* A portable thin NaI(Tl) detector connected to a single-channel analyzer with count rate meter (FIDLER) was tested at Haddam Neck and proved to be a sensitive detector under ambient conditions of the low-energy photons emitted by ^{133}Xe.(4) The hand-held detector consists of a cylindrical 13-cm-dia. x 1.6-mm-thick crystal optically coupled through a quartz light pipe to a 13-cm-dia. photomultiplier tube. The pulse rate meter (Eberline Instr. Co. Model PRM-5–3) is battery powered and contains a linear-log readout meter and three independent energy discrimination settings. Each of the three provides single-channel pulse-height analyzer capability. Before use, the meter was attached to a multichannel analyzer and one energy setting was adjusted to the center of the 81-keV spectral peak with a range corresponding to one half-width. The gamma-ray spectrum of ^{133}Xe in the 10 to 160 keV energy range with this arrangement is shown in Figure 6.2.

The scintillation detector was calibrated for 81 keV photon response by means of 400 µCi of ^{133}Xe in a 3-cc glass vial. Count rate meter readings were related to exposure rates by calibrating at the same time with a Shonka muscle-equivalent ionization chamber and electrometer used for low level radiation measurements. The relationship between count and gamma-ray exposure rates was linear for the source located at three distances from the two instruments. A counting rate with the scintillation detector of 15,000 counts/min was equivalent to 1 µR/h, i.e., 9 x 10^5 counts = 1 µR.

One survey meter was connected to a strip chart recorder (Texas Instrument Servo/Writer #2, Model PSO-W/6A), to plot readings continuously at a chart speed of 2.5 cm/min. Meter and chart-recorder performances were tested periodically during field operation with a 10.7-yr ^{133}Ba source, whose gamma rays include one of 81 keV energy.

6.2.3 *Measurements.* Three field tests were conducted 0.5 to 0.6 km distant from the stack at the locations shown in Figure 6.1. Atmospheric stability conditions from slightly unstable to neutral were selected so that the plume was likely to be measurable at ground level. The scintillation detectors were used to locate the plume centerline. Three or four compressed air samples were collected in each test. Background air samples were obtained upwind of the plant during tests 1 and 2.

The first test was conducted on September 16, 1970, from 1227 to 1429 hours, approximately 0.55 km WNW of the stack on the north side of the plant access road. The ^{133}Xe concentration in the surge sphere was 1.1 µCi/cc (see Table 3.2). Three air samples were obtained but ^{133}Xe was found only in the third. Because the winds shifted constantly, results could not be interpreted in terms of dispersion factors. The usefulness of the scintillation detectors for detecting the

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*We thank Mr. Sam Cummings of Eastern Environmental Radiation Facility, EPA, for performing these analyses.

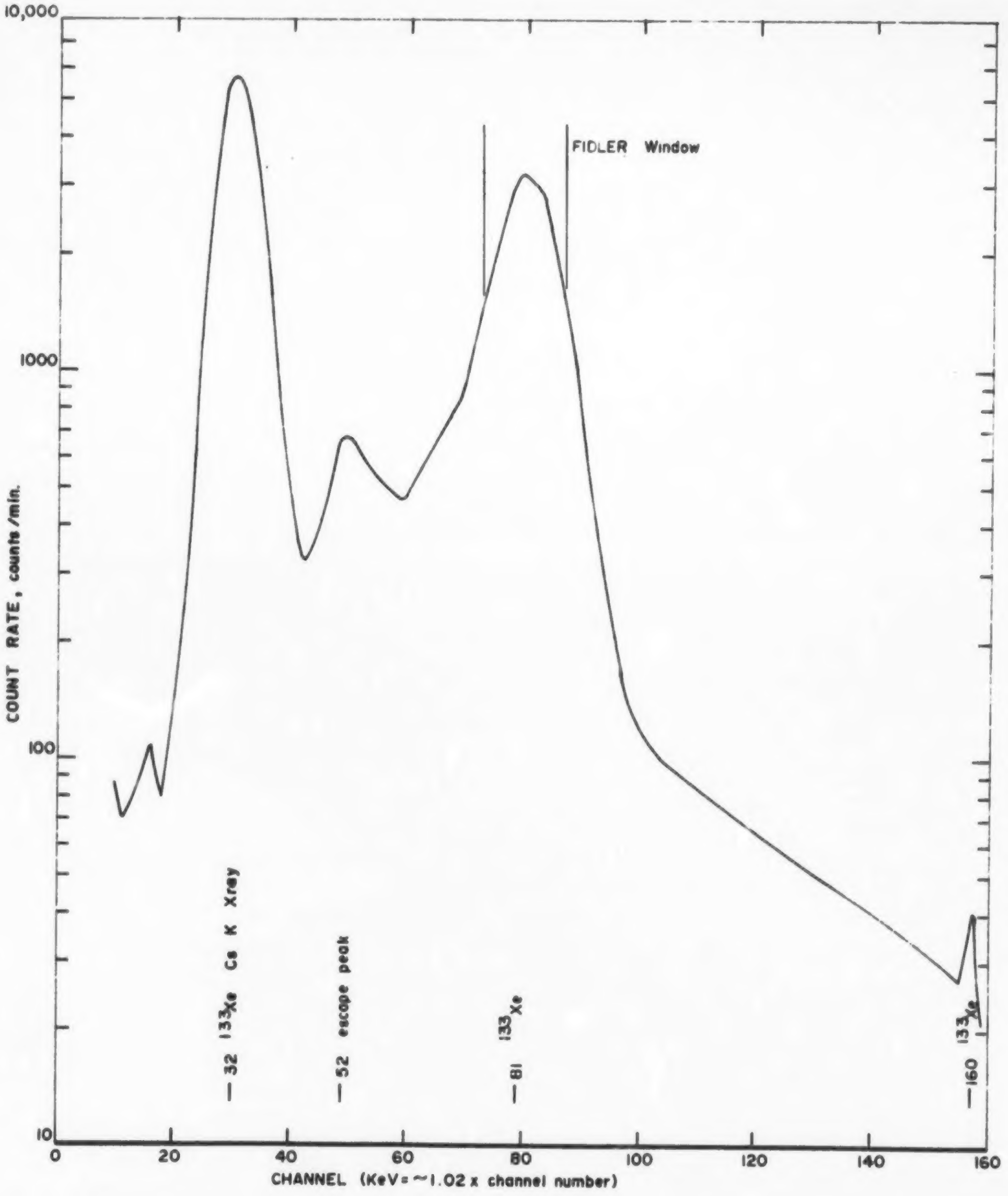

Figure 6.2 Gamma-ray spectrum of waste surge sphere gas.
Detector: 1.6 mm x 12.7 cm dia NaI(Tl) scintillation detector.
Sample: 9 cc bottle of gas collected 1005 hrs, March 15, 1971
Count: 1 min at 1710 hrs, March 15, 1971

plume was demonstrated, and ^{133}Xe concentrations at ground level near the station boundary were found to be detectable while the sphere was being discharged under these conditions.

The second test, on March 15, 1971, from 1339 to 1615 hours, did not yield useful results. Sampling was conducted 0.55 km NW of the stack on the hillside north of the access road. Traverses of the hillside from WNW to ENE with scintillation detectors showed only background values of 2,700 to 3,200 counts/min in the 81 keV channel The ^{133}Xe concentration in the surge sphere was 7.2 x 10^{-2} µCi/cc. The sample of stack effluent for determining ^{133}Xe discharge rates was not satisfactory. It was determined later that the discharged ^{133}Xe was at levels too low to be detected in the environment with the survey meters.

During the third test, on April 16, 1971, a sampling station was located on site 0.6 km ESE of the stack at the same elevation as the plant grounds. The location was bounded on the NE by the discharge canal and beyond that by a steep hillside. The sky was overcast in advance of an air mass moving toward the southeast. Wind velocities averaged for 10 min intervals varied from 2 to 8 m/s, with gusts to 11 m/s. Variations in wind direction for 10 min averages ranged from ± 10° to ± 80°, with an overall average of ± 45°. The temperature was 7.5° C (45.5° F) at the surface and 6.7° C (44° F) at the top of the tower. The temperature gradient, overcast skies, and fast winds in very irregular terrain characterize the atmosphere as slightly unstable (Pasquill Category C).(5)

Table 6.1 summarizes the test conditions on April 16. Gas from the surge sphere was released from 0845 to 1200 hours. Krypton-85 and ^{133}Xe emission rates were determined from a stack effluent gas sample taken shortly after the release started (see Section 3.1.3). At 1100 hours, the release rate from the sphere was increased from 1,890 cc/s to 3,300 cc/s. Release rates for samples No. 3 and 4 in Table 6.2 were computed from the ratio of these flow rates and, in the case of sample No. 3, the fraction of time during the flow rates. Sampling was twice suspended briefly when it was evident that the plume had shifted significantly. After 1135 hrs, the plume was no longer detected due to a major wind shift.

6.2.4 *Estimation of atmospheric dispersion* Atmospheric dispersion along the plume centerline at the sampling point on April 16 was estimated by the Pasquill-Gifford dispersion equation(5) in Appendix D.3. Vertical and horizontal plume dispersion coefficients apply for slight atmospheric instability. These coefficients, however, are for open and level ground and sampling times of about 10 min. Calculations of plume rise above the stack were based on ASME recommendations,(6) also given in Appendix D.3. These atmospheric dispersion values are listed in Table 6.2.

6.2.5 *Results and discussion.* Xenon-133 was measured in all air samples collected during the test at levels well above the limits of detection, as shown in Table 6.2. The highest concentration was in sample No. 2. Krypton-85 was found in both samples analyzed for that radionuclide, at the same concentration relative to ^{133}Xe as in the sample from the primary vent stack (see Table 3.8).

The atmospheric dispersion values obtained by dividing the measured concentrations in ground-level air by the stack release rates in Table 6.1 agreed with values for the centerline of the plume calculated by the Pasquill-Gifford technique for samples No. 1, 3, and 4. The higher measured concentration value in sample 2 was confirmed by the elevated radiation exposure rates observed with the survey meters (see Table 6.2). No

Table 6.1

Test Conditions for Sampling Haddam Neck Stack Effluents at Ground Level on Site, April 16, 1971

	Sample number			
	1	2	3	4
Sampling interval, hrs.	0906–0942	0951–1029*	1040–1118**	1120–1154
Duration of plume detection, min	15	35	31	16
Average wind speed, m/s	3.2	5.0	5.1	5.4
Stack release rate, µCi/s				
^{85}Kr	290	290	445	500
^{133}Xe	1700	1700	2600	2900

* except 1003–1005

**except 1050–1055

Table 6.2

Airborne Radionuclide Concentrations and Radiation Exposure Rates Measured at Ground Level on Site During Waste Surge Sphere Release, April 16, 1971

	Sample number			
	1	2	3	4
Air sample volume, m^3	0.47	0.47	0.44	0.47
Measured concentration, µCi/m^3				
^{85}Kr	NA	$5.5 \pm 0.1 \times 10^{-3}$	NA	$2.9 \pm 0.1 \times 10^{-3}$
^{133}Xe	$1.4 \pm 0.1 \times 10^{-2}$	$3.5 \pm 0.1 \times 10^{-2}$	$1.6 \pm 0.1 \times 10^{-2}$	$1.7 \pm 0.1 \times 10^{-2}$
Atmospheric dispersion (χ/Q), s/m^3				
Measured				
^{85}Kr	—	1.9×10^{-5}	—	5.7×10^{-4}
^{133}Xe	8.1×10^{-4}	2.0×10^{-5}	6.3×10^{-4}	5.7×10^{-4}
Calculated	8.8×10^{-4}	7.3×10^{-4}	7.1×10^{-4}	7.0×10^{-4}
Total net counts (81 keV)	24,000	72,000	46,000	34,000
Radiation exposure rate*, µR/hr	0.11	0.14	0.10	0.14

* Normalized to hourly rates, based on duration of plume detection

± values are analytical precision at 2-sigma

NA-not analyzed

reason can be given for the reduced dispersion during this period.

Ground-level air concentrations computed with the Haddam Neck diffusion model given in Appendices D.1 and D.2 were higher than sampling results, from slightly higher for sample No. 2 to more than 5-fold for No. 1. The relative concentration value normalized for wind speed ($\chi \bar{u}/Q$) of 1.4×10^{-4} m^{-2} for a 0.6-km distance from the stack, from Appendix D.2, at the average wind speeds given in Table 6.1 results in dispersion values (χ/Q) from 2.6×10^{-5} to 4.5×10^{-5} s/m^3. The computed dispersion is increased four-fold by including the factor for elevated discharges.

The plume was observed with the scintillation survey meters during the entire test on April 16, 1971. The count rates given in Figure 6.3 indicate the extent to which radiation levels fluctuated at the air sampling location. The net counts in Table 6.2 were calculated by determining the mean count rate less background in each 10-second chart interval, averaging the 6 values for each minute, and then integrating the indicated periods. Background activity remained constant at approximately 2,500 counts/min. In general, the variations in net counts were proportional to the measured ^{133}Xe concentrations, in that 1000 counts corresponded to 5×10^{-4} pCi/m^3.

The average gamma-ray exposure rate due to the plume of stack gas while it was overhead was approximately 0.13 µR/h (see Table 6.2). The rates were derived from the scintillation detector counting results given in the table by applying the conversion factor indicated in Section 6.2.2 and adjusting for the duration of plume detection during each sampling interval (see Table 6.1). Thus, for the first sample:

$$(2.4 \times 10^{4} \text{ counts}/0.25 \text{ h})/(9 \times 10^{5} \text{ counts}/\mu\text{R}) = 0.11 \ \mu\text{R/h}$$

The average gamma-ray exposure rate from ^{133}Xe of 0.13 µR/h, at the average ^{133}Xe stack release rate of 2,200 µCi/s during the period of plume detection, yields a ratio of 5.8×10^{-5} µR/h per µCi/s, or 1 µR per 61 Ci ^{133}Xe. A discharge of 2,000 Ci ^{133}Xe per year would thus result in an annual gamma-ray exposure of *33 µR to an individual submerged continuously in the* plume 0.6 km distant from the stack under similar atmospheric conditions. The annual dose from all the

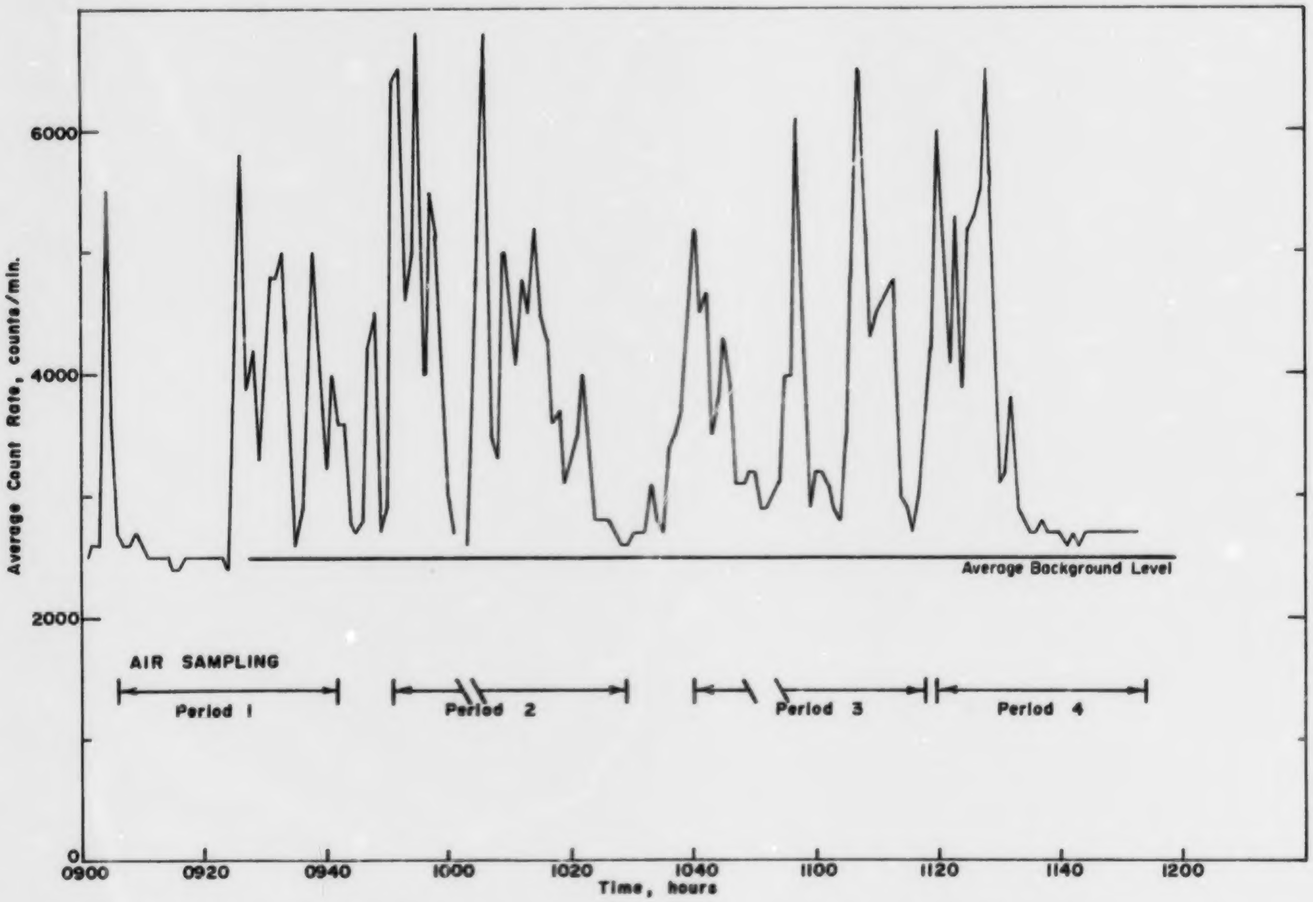

Figure 6.3 Scintillation detector response during air sampling on April 16, 1971. (Count rates are averaged for one-minute periods)

radiations of ^{133}Xe, including conversion electrons, beta particles, and X-rays, would be 0.2 mrem because the gamma rays contribute only 17 percent of the dose.(1) This is consistent with the annual dose of 0.3 mrem given in Section 3.3.13 for the nearest habitation, 0.7 km WNW, based on measured discharges and the station operator's meteorological model.

The one ground level air sample showing positive results in the test on September 16, 1970, contained ^{133}Xe at a concentration of $4.1 \pm 0.2 \times 10^{-3}$ µCi/m^3. Thus, because of the variability in wind direction, the ^{133}Xe concentration in a sample collected at approximately the same discharge rate (see Table 3.8) and distance was 5-fold lower than on April 16. Most radiation exposure rates measured with the scintillation detector on September 16 were at background values, but a few brief elevated readings at location 1 (see Figure 6.1) showed net values as high as 4,200 counts/min (0.3 µR/h) due to the stack release. The following brief exposure rates from the plume were also observed on September 16 at the locations shown in Figure 6.1:

Location	Distance, km	Brief net exposure rate, µR/h
1a	1.3	0.04
1b	1.4	0.07
1c	1.7	0.17
1d	1.7	0.25
1e	1.7	0.02

No elevated count rates were observed to the east or west of the above locations, or at greater distances from the stack.

The tests demonstrate that, under some conditions, 1) a portable scintillation detector (FIDLER) can be used to measure directly even the relatively low radioactivity levels discharged from PWR stacks and 2) ^{133}Xe and ^{85}Kr can be measured in samples of ground-level air. Under meteorological conditions leading to the observed dispersion factors, the scintillation detector can detect, at the fence-line, ^{133}Xe releases as low as 1,000 µCi/s. The ^{133}Xe analysis is approximately an order of magnitude more sensitive.

6.3 References

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7. RADIONUCLIDES AND RADIATION IN THE TERRESTRIAL ENVIRONMENT

7.1 Introduction

7.1.1 *Sampling.* Release data by the Haddam Neck station (see Appendices B.2-B.4) and radioactivity measurements during this study of airborne effluents (Section 3) and ground-level air in the immediate environment (Section 6) suggest that radionuclide concentrations in ground-level air and deposition on ground and vegetation due to station operation were very low. Environmental samples are analyzed for radioactivity by the Connecticut Department of Environmental Protection (CDEP) and the station's contractor for environmental surveillance.(1,2) During 1970 and 1971, CDEP performed gross alpha, gross beta, and some gamma-ray spectral analyses in potable well water, vegetation, milk and fodder (hay and silage) and measured ^{89}Sr and ^{90}Sr in milk and ^{3}H in well water. The station's contractor measured gross beta activity in airborne particles; gross alpha and beta activity, gamma-ray emitters and ^{3}H in well water; gross beta activity and gamma-ray emitters in vegetation; and external radiation exposure with thermoluminescent dosimeters (TLD's).

The following samples and measurements were obtained in the neighborhood of the Haddam Neck station (see Figure 7.1):

(1) Four samples of potable well water were collected for ^{3}H analysis on August 6, 1970, from homes located 0.8 to 1.5 km from the station. A control sample was obtained from a home 4.6 km from the station.

(2) Three food crops—blueberries, lettuce plus cabbage leaves (the control sample contained only lettuce), and sweet corn—were collected near the station and at background locations for radionuclide analysis.

(3) Milk samples were provided for radionuclide analysis by CDEP from three dairy farms located 2 to 4 km from the station, and from a 9.5-km-distant dairy farm as control.

(4) Bovine thyroids for ^{131}I analysis were collected at slaughter from two nearby cows that had grazed on a pasture 3.3 km distant from the station. Control thyroids were obtained from cattle that had grazed 28 and 18 km distant.

(5) Tissue of two deer killed near Haddam Neck and two deer killed at distant locations were compared for radionuclide content.

(6) External radiation exposure was measured with NaI(Tl) survey meters at 17 points on-site and at 19 points off-site within 5 km of the station.

None of the radionuclides found in well water, milk, food, or deer tissue is attributed to Haddam Neck station, as described in detail in Sections 7.2, 7.3, 7.4, and 7.6. The only radionuclide believed to be from the station was ^{131}I found in one cow's thyroid. Calculations of expected concentrations of ^{131}I in bovine thyroids and milk are presented in Section 7.5.2 and Appendices D.4 and D.5. Although grossly approximate, they demonstrate the procedure, indicate the magnitude of radionuclide concentrations that may be attributed to Haddam Neck station, and illustrate the higher sensitivity of measuring ^{131}I in bovine thyroid than in milk. The calculations are based on ^{131}I values measured in airborne effluents (Section 3.3) and meteorological data from the nearest station of the National Oceanic and Atmospheric Administration (NOAA).

The external radiation exposure rate above background was estimated to be about 1 microroentgen per hour (μR/h) at a location on the Haddam Neck station exclusion perimeter, and 0.3 μR/h or less at the nearest residences, as described in Section 7.7. This radiation was attributed to gamma rays from radioactive waste stored at the station.

7.1.2 *Environment of Haddam Neck.*(3,4) The site consists of 223 hectare (525 acres) and is bounded by the Salmon River to the east and the Connecticut River to the south and west (See Figure 6.1). The elevation of the plant site is about 3 m (10 ft) at the river and 6.5 m (21 ft) at the plant yard. Heavily wooded hills, cresting near 90 m MSL, rise steeply just beyond the northeast

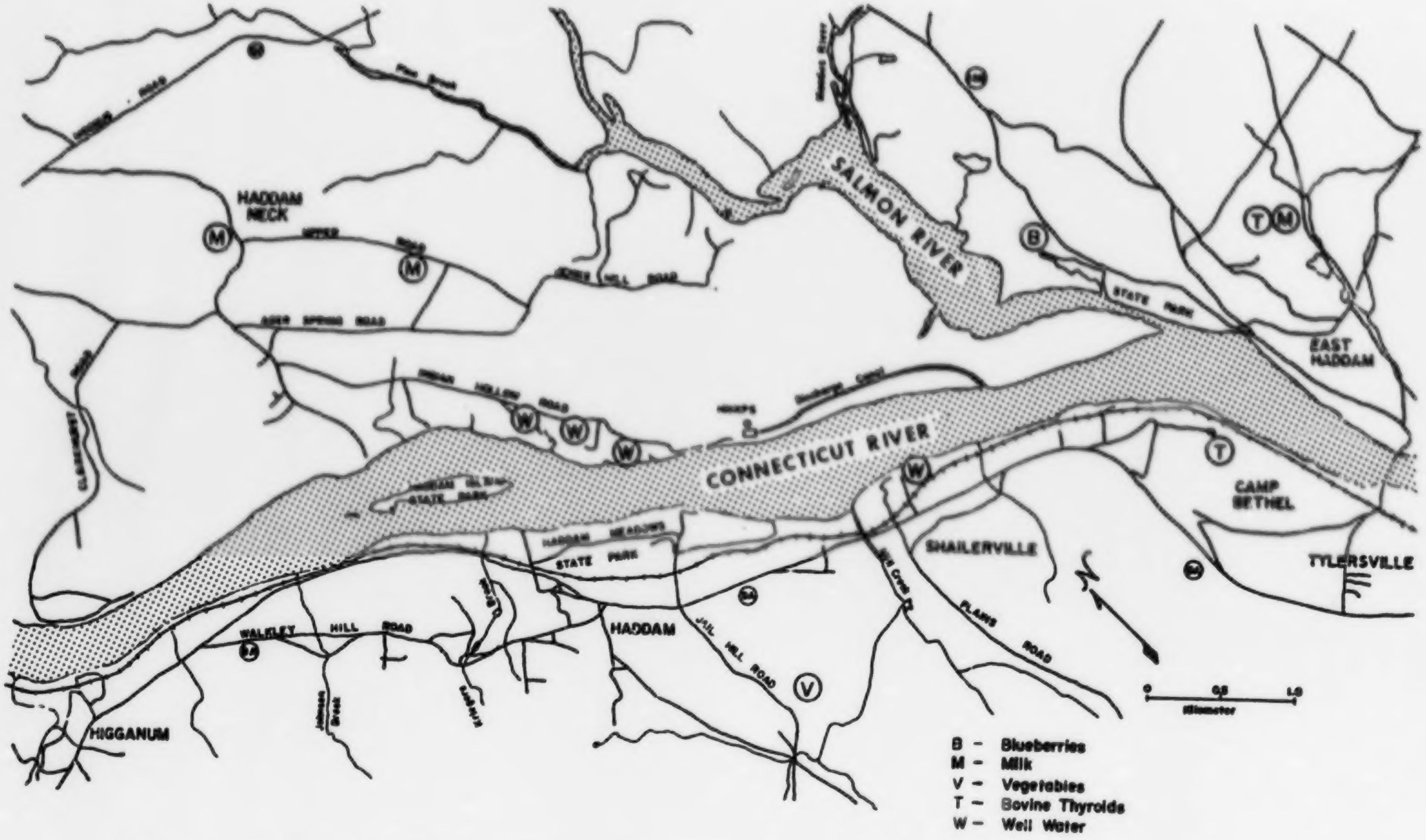
HADDAM NECK
UPPER ROAD
SKINNER SPRING ROAD
JONES HILL ROAD
INDIAN HOLLOW ROAD
Pine Brook
SALMON RIVER
STATE PARK
EAST HADDAM
CONNECTICUT RIVER
CAMP BETHEL
TYLERSVILLE
SHAILERVILLE
PLAINS ROAD
HADDAM MEADOWS
STATE PARK
HADDAM
JAIL HILL ROAD
WALKLEY HILL ROAD
HIGGANUM
Johnson Brook
CLARKMONT
0
0.5
1.0
Kilometer
N
B
M
V
T
W
B - Blueberries
M - Milk
V - Vegetables
T - Bovine Thyroids
W - Well Water

Figure 7.1 Terrestrial Sampling Locations

perimeter of the yard fence. The elevation of the hills approaches 120 m northwest of the plant site.

The plant is located in a region with a relatively low population density. Haddam, about 1.6 km (1 mile) west across the river, is the nearest community and has a population of 130.* The largest nearby town is Middletown, 16 km (10 miles) northwest of the site, with a population of 37,000. The rural nature of the site vicinity is suggested below by a population tabulation:(4)

Distance from site (miles)†	Accumulated population
1/2	11
1	278
5	8,550
10	55,239
15	160,055
20	481,111
25	1,187,352
30	1,697,308
40	2,292,932
50	2,937,764

†1 mile = 1.61 km

The nearby directional population within 3 km of the site was:(4)**

Direction	Population	Percent
N	98	5
NNE	194	10
NE	128	7
ENE	220	11
E	51	3
ESE	32	2
SE	134	7
SSE	131	7
S	92	5
SSW	108	6
SW	152	8
WSW	129	7
W	288	15
WNW	15	1
NW	79	4
NNW	86	4

The population near the Connecticut River fluctuates because of summer homes and riverside cottages in the area. Several small resorts are located near East Haddam, 5 km east of the plant.

The nearest dairy farms in the area are located 2.6 km (1.6 miles) NW and 3.3 km (2.1 miles) ESE of the site. Other dairy farms in the general area are located near East Hampton, Killingworth, and Middletown. Some goat milk is also reported to be produced in the area. Hay and silage are harvested in the local area; the fields nearest to the station are at East Haddam. A few small private gardens, but no truck farms, are located in the vicinity of the station. The Connecticut River near and below the station is not used as a public water supply; drinking water for local residents is taken from wells. The nearest wells are located at private homes along Indian Hollow Road, the access road to the station. Except for UARCO, Inc. at Deep River, 11 km (7 miles) SE, the nearest industry is located near Middletown.

7.2 Tritium in Well Water

7.2.1 *Sampling and analysis.* Duplicate samples of well water were collected at five private homes on August 6, 1970 (see Figure 7.1). Three of the samples were from the homes nearest the station along Indian Hollow Road, 0.8, 1.1 and 1.5 km NW of the station. The fourth sample was from a well about 10 m from the far bank of the Connecticut River, 1.2 km SE of the station. The fifth was a control sample, from a well on high ground 4.6 km ESE of the station. The samples were divided and analyzed for tritium by liquid scintillation techniques at this laboratory(5) and by CDEP.

7.2.2 *Results and discussion.* All well water samples contained less than 0.8 pCi/ml of tritium, the 3-sigma detection limit. The CDEP reported similar values (0.6 to <0.3 pCi/ml). The only elevated ^{3}H values in this type of sample had been found in one well, located near the discharge canal on station property, which contained approximately 4 pCi/ml according to CDEP(1) and the station's contractor.(2) Gross radioactivity levels in well water samples from homes nearby and at background locations were reported to be 5 to < 1 pCi/liter for alpha and 5 to about 0.5 pCi/liter for beta.(1,2) These data suggest that diffusion, if any, of radioactivity from the discharge canal or river into the water table supplying off-site wells was below the limit of detection.

The upper limit of the radiation dose from drinking ^{3}H in well water can be estimated from Appendix E.2. If the tritium concentration in the well water had been at the detection limit of 0.8 pCi/ml, an individual drinking one liter of water per day would have received a dose to the total body of 0.06 mrem/yr.(6)

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*Populations are from the 1970 census.

**The population in this area is 1937 and the sum of the percentages is 102 due to rounding off.

7.3 Radionuclides in Food Crops

7.3.1 *Sampling and analysis.** Collecting an adequate quantity of vegetables for analysis was difficult because truck farming does not exist near the station. Small quantities of home-grown produce are available at roadside stands, although some items for sale are brought from distant farms.

On August 6, 1970, fresh blueberries were obtained from a field near the Salmon River, 2.3 km east of the station, at an elevation of 15 m MSL. On the same date, sweet corn and lettuce mixed with some cabbage leaves were collected from a private garden 1.8 km SW of the station near the top of the ridge, about 75 m MSL (see Figure 7.1). Because of the late date, the lettuce plants were in poor condition and approximately an equal weight of immature cabbage plants had to be added to obtain a sufficient sample for analysis. Control samples of blueberries, lettuce and sweet corn that had been harvested August 5, 1970, at farms 22–34 km NW of the station were obtained at vegetable stands. No cabbage was available from regional truck farms at that time.

The corn was husked and the kernels were removed from the ears. Prior to analysis, all samples were washed with tap water, as for food preparation. The corn husks were also analyzed to ascertain possible airborne deposition on their surfaces. Samples in both fresh and ashed (400° C) states were analyzed with a 10– x 10-cm NaI(Tl) detector and multichannel analyzer for radionuclides that emit gamma rays, and the ash was analyzed for ^{90}Sr.(5) Tritium and ^{14}C were determined by treating samples in a combustion train, collecting water and CO_2, and measuring the radioactivity with liquid scintillation and gas counting techniques.**

7.3.2 *Results and discussion.* No photon-emitting radionuclides attributable to the station were detected in any of the samples. Only naturally occurring ^{40}K, ^{226}Ra and ^{232}Th and traces of the fallout nuclides ^{137}Cs, ^{95}Zr and ^{95}Nb were detected. As shown in Table 7.1, there were no significant differences in the content of ^{3}H and ^{14}C in samples collected near the station and in those from a distance, or, with one exception, in the content of ^{90}Sr. The ^{90}Sr content was higher in the nearby sample of mixed lettuce and cabbage than in the control sample of lettuce, but this may be due to a higher ^{90}Sr content in the cabbage. The ^{14}C specific activity was consistently higher than in the aquatic background samples (Tables 5.4 and 5.6).

Analyses of nearby vegetation (weeds, grass, hay, silage) in 1970–1971 show only ^{40}K and traces of the fallout radionuclides ^{95}Zr-^{95}Nb, ^{103}Ru, ^{106}Ru, ^{137}Cs, ^{140}Ba-^{140}La and ^{144}Ce.(1,2) The average gross beta activity of these samples was reported to be about 30 pCi/g ash,(2) similar to that of the controls.

Table 7.1

Radionuclides in Food Collected August 6, 1970

Sample	Distance	^{3}H, pCi/ml H_2O	^{14}C, pCi/g C	^{90}Sr, pCi/g ash
Blueberries, near	2.3 km E	2.6 ± 0.6	7.8 ± 0.2	6.6 ± 0.3
Blueberries, distant	22 km NNW	1.2 ± 0.6	7.8 ± 0.2	5.9 ± 0.3
Lettuce + Cabbage near	1.8 km SW	NA	NA	5.5 ± 0.1
Lettuce, distant	34 km NNW	NA	NA	1.5 ± 0.2
Corn kernels, near	1.8 km SW	<0.9	7.4 ± 0.2	0.64 ± 0.05
Corn kernels, distant	27 km NNW	<0.7	7.9 ± 0.2	0.53 ± 0.04
Corn husks, near	1.8 km SW	<0.7	8.5 ± 0.2	1.4 ± 0.1
Corn husks, distant	27 km NNW	<0.9	8.1 ± 0.2	3.7 ± 0.1

± values are 2-sigma of counting error; < values are 3-sigma error.
NA – not analyzed.

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*We thank Messrs. E. W. Prout and G. G. Curtis, University of Connecticut Cooperative Extension Service, Haddam, Connecticut, for their advice on regional agricultural practices.

**We thank E. J. Troianello, EPA, Winchester, Massachusetts, for the ^{14}C and ^{3}H analyses.

7.4 Radionuclides in Milk

7.4.1 *Sampling and analysis.* On August 4, 1970, raw milk samples were collected from the dairy farms described in Table 7.2, at locations shown in Figure 7.1. Photon-emitting radionuclides were measured by analyzing 3.5 liters of milk with a 10– x 10-cm NaI(Tl) detector, and ^{89}Sr and ^{90}Sr were determined by radiochemical analysis.(5) Tritium and ^{14}C were determined as described in Section 7.3.1.

7.4.2 *Results and discussion.* The results in Table 7.2 show that ^{3}H, ^{131}I, and ^{140}Ba were below the minimum detectable level in all samples. The concentrations of ^{14}C, ^{89}Sr, ^{90}Sr and ^{137}Cs in milk from the nearby dairy farms and the control dairy farm were similar, except for a higher ^{90}Sr value in one nearby sample and a higher ^{137}Cs value in the control sample.

These results are similar to data tabulated below for average radionuclide concentrations in milk. The CDEP reported on six dairies located in the Haddam area, and the U. S. Public Health Service Pasteurized Milk Network (PMN)(1,7) reported on milk sold in Hartford:

Radionuclide	CDEP 1970 Avg.,(1) pCi/liter	PMN July and Aug 1970 Avg.,(7) pCi/liter
^{89}Sr	<3	<5
^{90}Sr	10	8
^{131}I	<10	<10
^{137}Cs	22	22
^{140}Ba	<10	<10

The concentrations of ^{90}Sr and ^{137}Cs were in the range reported at PMN stations throughout the U.S.; these concentrations were attributed to fallout from nuclear tests in the atmosphere.(7)

7.5 Iodine-131 in Bovine Thyroids

7.5.1 *Sampling and analysis.* Thyroids from cattle grazing near a nuclear power station can be sensitive qualitative indicators of the presence of ^{131}I in the environment.(8) They may be of particular value for estimating ^{131}I levels when milk samples cannot be procured but beef cattle graze nearby, or when predicted concentrations in milk are below the detection limit.

One thyroid was obtained from a cow at slaughter on September 16, 1970. The cow had grazed on a hill 3.3 km ESE of the station (see Figure 7.1) for several weeks prior to removal from pasture on September 14. It had received a daily supplement of 5 kg of grain, 2 kg of prepared hay, and a variable amount of fresh-cut corn. Control thyroids were obtained at slaughter on September 16 from two cows from Wallingford, Connecticut, 28 km WSW of the station. These control cows were on open pasture and had received an undetermined amount of grain supplement.

A second thyroid was obtained on June 3, 1971, from a cow that had grazed on open pasture in a low-lying meadow across the Connecticut River 3.3 km SE of the station. This animal had received some daily grain supplement but otherwise consumed grass almost exclusively for several weeks until its slaughter. Control thyroids were collected on June 17, 1971, from two oxen that were pastured 18 km east of the station in Middlesex, Connecticut. The oxen were reported to have eaten mainly fresh pasture grass and a variable amount of stored hay.

Thyroids were excised by veterinarians* and shipped to the laboratory preserved with powdered

Table 7.2

Radionuclide Concentrations in Milk Samples Collected August 4, 1970*

Distance	^{3}H,** pCi/l	^{14}C,† pCi/g C	^{90}Sr, pCi/l	^{89}Sr, pCi/l	^{131}I, pCi/l	^{140}Ba, pCi/l	^{137}Cs, pCi/l
3.7 km NNW	<800	7.6±0.3	16 ± 1	3 ± 2	<3	<3	19 ± 2
2.6 km NNW	<800	6.4±0.3	10 ± 1	3 ± 2	<3	<3	26 ± 2
3.3 km ESE	<800	6.7±0.3	12 ± 1	<3	<3	<3	20 ± 2
9.5 km NW	<800	7.1±0.3	12 ± 1	4 ± 2	<3	<3	32 ± 2

* ± values are 2σ counting error; < values are 3σ.

** Tritium by combustion (1000g milk = 970g H_2O).

† Average of 5.5% carbon found experimentally.

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*We thank Drs. M. Ross, W. Stanhaites and A. Nigro, Meat Inspection Program, U. S. Department of Agriculture, Dr. O. Burr, Chester, Connecticut, Mr. J. Smolen, Connecticut Dept. of Environmental Protection, and Mr. H. Diehl, Connecticut Department of Agriculture and Natural Resources, for helping to procure these samples.

paraformaldehyde in plastic bags. The samples were dissected into small pieces, placed in 9-cm-dia. x 1.5-cm plastic Petri dishes and counted between two 10- x 10-cm NaI(Tl) detectors within an annular NaI(Tl) anticoincidence shield. The detection limit, based on the 3-sigma deviation observed in a series of background counts was 2 to 5 pCi per thyroid when adjusted for radioactive decay.

7.5.2 *Predicted concentration in bovine thyroids.* Iodine-131 levels in cows' thyroids and milk were predicted from release rates at the station stack, the station's model for dispersion in air,(9) and reported values for iodine deposition velocity and metabolic transfer factors for the cow. The dry deposition, D, in pCi/m^2, (it did not rain while the cows were on pasture just before slaughter) was computed by:

$$D = Q'_o(T/\bar{u})(\chi\bar{u}/Q)v_d(\chi_L/\chi) \qquad (7.1)$$

where:

Q'_o = release rate of ^{131}I at the stack during the period of interest, pCi/s (averaged from Table 3.11)

T = duration of wind toward sampling location, s

$\bar{u}$ = velocity of wind toward sampling location, m/s

$\chi\bar{u}/Q$ = relative concentration normalized for wind velocity, m^{-2} (from Appendix D.2)

v_d = deposition velocity of ^{131}I on grass, m/s

χ_L/χ = ratio of average long-term concentration in sector ground-level air to that at centerline.

The values of T and $\bar{u}$ were obtained from climatological data at the NOAH meteorological station at Bradley Field, Hartford.(10) To simplify the wind frequency calculations, a 90° sector (270° to 360°) was used, allowing for channeling effects of the valley while also encompassing both pasture locations, but the duration value was then divided by 4 to apply the exposure to a 22.5° sector. A deposition velocity of 0.01 m/s was used.(11,12) The ratio χ_L/χ was taken to be 0.5, based on the discussion in Appendix D.1. The average value of $\chi\bar{u}/Q$ at a 3.3-km distance is 2.5 x 10^{-5} m^{-2} according to Appendix D.2. The ^{131}I in daily deposition was computed to decay with a 5-day environmental half-life(14) to estimate day-by-day concentrations of ^{131}I on grass shown in Appendix D.4.

The net cumulative concentration of ^{131}I in the two thyroids was calculated by assuming that each cow grazed effectively 45 m^2 of pasture each day,(13) and that 20 percent of the iodine intake was taken up by the thyroid, where it decayed with an effective half-life of 7 days.(13) These calculations are shown in Appendix D.5 and the results, in Table 7.3.

Although no milk was collected from these cows at this time, hypothetical concentrations of ^{131}I in the milk were computed in Appendix D.5 for a series of single doses according to the graph by Garner and Russell.(13) Thyroid and milk detection sensitivities for ^{131}I are compared for these circumstances in Table 7.3. The ratios of ^{131}I per liter of milk to ^{131}I per gram of thyroid, at thyroid weights of 50 g, are less than the reported ratio of 1:12.(15)

7.5.3 *Results and discussion.* Iodine-131 attributable to reactor operation was detected only in the thyroid of the cow that had grazed 3.3 km ESE of the station until September 14, 1970 (see Table 7.4). The measured ^{131}I burden in this thyroid was 21 ± 3 (2-sigma) pCi, or 0.44 pCi/g. The level predicted from the station diffusion model was six times as great (see Table 7.3). The undetectably low ^{131}I content of the thyroid collected June 17, 1971, was also less than the predicted value. The ^{131}I concentration would be overestimated because the elevation of the discharge point was not considered (see Section 6.2.5). Another reason for the discrepancy is the use of meteorological data from the distant and dissimilar location at Hartford. The computed concentrations of ^{131}I in milk from these cows were well below the detection limits shown in Table 7.2.

According to the Federal Radiation Council, a daily intake of 80 pCi ^{131}I will result in a dose to a child's thyroid of 500 mrem/yr.(16) At a daily intake of 1 liter of milk with the estimated ^{131}I concentration of 0.44 x 1/12 = 0.04 pCi/liter, the dose to the thyroid would be 0.2 mrem/yr.

7.6 Radionuclides in Deer

7.6.1 *Sampling and analysis.* To begin evaluation of the radionuclide content in wildlife, four white-tail deer *(Odocoileus virginianus)* were collected, two within 5 km of the reactor, and two at a distance of 21 km. The deer are described in Table 7.5. Deer D-1 and D-2 were killed in automobile accidents on Highway 2, D-4 was also killed in an automobile accident at the intersection of Highways 196 and 151, and D-3 was shot at Haddam Neck. Samples of bone (femur), muscle, liver and kidney were preserved in plastic bags on dry ice. No bone sample was supplied for deer D-3, and, although requested, rumen content—a direct indicator of recent radionuclide intake with food—was not collected from any deer. Muscle, liver and kidney samples were ashed at 400°C and analyzed for photon-emitting radionuclides by spectrometry with a 10- x 10-cm

Table 7.3

Estimated Levels of ^{131}I in Cow Thyroids

Location/ collection period	Assumed Q_o'* pCi/sec	Thyroid level,** pCi		Hypothetical milk concentration, pCi/1		Concentration ratio thyroid/milk,† pCi/g per pCi/liter
		Generalized	Q_o' as shown	Generalized	Q_o' as shown	
3.3 km ESE, Sept. 1970	820	0.16 Q_o'	130	7.8×10^{-5} Q_o'	6.4×10^{-2}	41
3.3 km SE, June 1971	58	0.062 Q_o'	3.6	7.6×10^{-5} Q_o'	4.4×10^{-3}	16

* The Sept. 1970 value is the average of 6 ^{131}I concentrations (Table 3.11) measured during continuous discharges in July-Sept. 1970, multiplied by the flow rate of 16.7 m^3/sec; the June 1971 value is the average of 3 ^{131}I concentrations (see footnote to Table 3.11) measured in June 1971, multiplied by 16.7 m^3/sec.

** $\chi \bar{u}/Q = 2.5 \times 10^{-5}$ m^{-2} (from Appendix D.2).

† A thyroid was assumed to weigh 50 g.

Table 7.4

^{131}I in Bovine Thyroids

Animal type	location	Last day on pasture	Thyroid weight, g	^{131}I content, pCi/thyroid*
Cow	3.3 km ESE	Sept. 14, 1970	49	21 ± 3
Cow	28 km WSW	Sept. 14, 1970	50	<3
Cow	28 km WSW	Sept. 14, 1970	33	<5
Cow	3.3 km SE	June 3, 1971	75	<2
Ox	18 km E	June 14, 1971	29	<2
Ox	18 km E	June 14, 1971	52	<2

Notes:

1. Thyroid samples contained variable amount of fat.
2. ± value based on 2-sigma of observed standard deviation in a series of background counts; < values are ± 3-sigma.

*Corrected for physical and biological decay of ^{131}I to last day on pasture.

Table 7.5

Description of Sampled Deer*

Deer No.	Location	Distance and direction from reactor, km	Date collected, 1970	Age, yrs	Sex
D-1	E. Gastonbury	21 NNW	Nov. 13	3.5	F
D-2	E. Gastonbury	21 NNW	Nov. 13	0.5	F
D-3	Haddam Neck	1.2 N	Aug. 15	2.5	M
D-4	Wopowog	4.8 NNW	Sept. 12	2.5	F

*We thank Edward Goldin, Game Biologist, Connecticut Board of Fisheries and Game, for collecting the deer and determining their ages.

NaI(Tl) detector or an 11-cm^3 Ge(Li) detector. Samples were also analyzed with a NaI(Tl) gamma-ray coincidence/anticoincidence system. Bone samples were ashed at 600° C. Bone and muscle samples were analyzed for radiostrontium by radiochemical analysis, and for stable calcium and strontium with an atomic absorption spectrophotometer.

7.6.2 *Results and discussion.* The only radionuclides detected in the deer samples were ^{90}Sr, ^{137}Cs, and naturally occurring ^{40}K, as shown in Table 7.6. The ^{58}Co and ^{60}Co content was found to be <40 pCi/kg each (3-sigma counting error) in liver and kidney samples. The average concentrations of ^{90}Sr and ^{137}Cs in the deer collected near the plant site were lower than those in the deer collected at a distance.

The average ^{137}Cs concentration in muscle was 2,400 ± 1,000 (± 1-sigma) pCi/kg or 740 ± 140 pCi/g potassium. No ^{134}Cs was detected; the minimum detectable concentration was 20 pCi/kg at the 3-sigma level. In the muscle, the average ^{90}Sr concentrations were 8 ± 1 pCi/kg, 43 ± 4 pCi/g calcium and 69 ± 14 pCi/mg strontium. The concentrations relative to calcium and strontium are similar to those in bone.

The ^{137}Cs concentrations in the muscle of these deer are somewhat higher than observed previously in deer collected in the vicinity of the Yankee-Rowe and Dresden nuclear power stations,(5,8) but the levels are not high compared to deer muscle collected at some areas distant from nuclear power stations.(17-19) Jenkins and Fendley reported numerous cases in which levels of ^{137}Cs in the muscle of white-tail deer from the southeastern United States approach 150,000 pCi/kg as a result of the concentration of fallout ^{137}Cs in certain types of vegetation.(17,19)

The average ^{90}Sr concentration in deer bone was 10,000 ± 8,000 pCi/kg, 62 ± 32 pCi/g calcium and 83 ± 54 pCi/mg strontium. These concentrations are similar to values observed in deer collected in 1969 from western Massachusetts,(5) and also to concentrations reported to be in deer from South Carolina, Colorado and California.(18,20-22)

The average ^{90}Sr concentration of 8 pCi/kg deer meat is only one-seventh of that reported for Alaskan caribou or reindeer meat,(23) but is about 8 times that in meat sampled as a typical component of New York City diets on three occasions in 1970.(24) The average stable and radiostrontium concentrations in deer muscle were approximately 1/1,200 of the concentration found in bones, similar to the ratio of 1/1,500 observed in deer samples collected from western Massachusetts in 1969.(5)

Because of the wide range of measured concentrations among samples, more samples would have to be collected and analyzed if radionuclide concentrations nearby and at a distance had to be compared with better precision. This problem of a wide range of concentration values is common among environmental samples, but is especially serious in animals whose radionuclide contents can be affected by their mobility.

7.6.3 *Estimated radiation dose from eating deer meat.* The radiation dose a person could receive from eating deer meat was estimated from daily intake-dose rate relationships as discussed in Section 5.5.5. According to these, a dose rate of 170 mrem/yr to the bone or whole body will result from a daily intake of 200 pCi ^{90}Sr or 15,000 pCi ^{137}Cs. At the average ^{90}Sr

Table 7.6

Radionuclide (pCi/kg)* and Stable Ion Concentration (g/kg)* in Deer Samples

Sample type	Nuclide	Distant D-1	Distant D-2	Nearby D-3	Nearby D-4
Muscle	^{137}Cs	3740 ± 30	2,030 ± 20	1320 ± 50	2550 ± 20
	^{90}Sr	8.4 ± 0.8	8.8 ± 0.8	6.0 ± 0.6	6.9 ± 0.6
	K	3.96	3.21	2.00	3.59
	Sr	0.00012	0.00010	0.00010	0.00012
	Ca	0.18	0.19	0.16	0.17
	ash wt./wet wt.	0.009	0.010	0.010	0.009
Bone	^{90}Sr	4300 ± 120	15,700 ± 400	NS	9600 ± 200
	Sr	0.075	0.108	NS	0.211
	Ca	117	159	NS	186
	ash wt./wet wt.	0.34	0.40	NS	0.47

* kg wet weight

Note: ± values are 2σ counting error.

NS – no sample was available.

concentration of 8 pCi/kg meat and a maximum annual consumption by an individual of 45 kg (0.12 kg/day),(25) the radiation dose to bone marrow is 0.8 mrem/yr. At the same consumption rate and an average ^{137}Cs concentration of 2,400 pCi/kg muscle, the radiation dose from ^{137}Cs to the whole body is 3.3 mrem/yr. To the average deer hunter who consumes an estimated 2.7 kg/yr deer meat,(25) the doses are 6 percent of those calculated above. These doses are believed to be from radionuclides in fallout, but provide an upper limit if some of the radioactivity in deer were from the station.

7.7 External Gamma Radiation

7.7.1 *Detection instruments.* Radiation exposure rates were measured with cylindrical NaI(Tl) gamma-ray detectors (5-cm diameter x 5-cm length) connected to portable count-rate meters. The instruments had been calibrated by comparing their count rates for gamma rays above 80 keV in the natural radiation background at Cincinnati with measurements by a muscle-equivalent ionization chamber and Shonka electrometer. Radiation levels during calibration ranged from 5 µR/h over water in a lake to 19 µR/h over granite. The count rate, C (in counts/min), of the survey instruments varied linearly with the radiation exposure rate, R (in µR/h), of the ionization chamber; a typical calibration curve had the equation $R = 7.0 \times 10^{-4}C + 3.3$. Radiation exposure rates at the measurement locations near Haddam Neck were computed by applying these calibration curves to the observed count rates.

Despite the dependence of the counting efficiency of the detectors on the energy distribution of the gamma-ray flux, the calibration curves had been found applicable in a variety of natural radiation backgrounds. In numerous measurements, the standard error of the survey meters was ± 0.35 µR/h, and the exposure values computed from the readings were within 4 percent of the values measured with the ionization chamber in 95 percent of the measurements.(26)

7.7.2 *Measurements.* The 36 radiation measurement locations shown in Figures 7.2 and 7.3 were selected for the following reasons:

(1) Point No. 1 was investigated as a source of high background radiation due to a large granite outcropping at a location remote from the station;

(2) Sixteen points, Nos. 2, 4-9 and 11–19, 0.5 km to 2.4 km distant from the center of the station, provided radiation exposures in the immediate environment of the station;

(3) Point Nos. 3 and 10 were considered to be sufficiently distant from Haddam Neck station but similar in natural radiation to yield terrestrial background values for comparing with and subtracting from exposure rates near the station;

(4) Seventeen points on site, but outside the security fence, Nos. 20–36, were intended to aid in identifying the source of external radiation from Haddam Neck station and to check off-site exposure values by extrapolating from these higher, more precise measurements.

Most exposure rates were measured on two occasions while the station was operating at full power. Detectors were held 1 m above ground surface. Count rates ranged from 5,000 to 100,000 counts/min.

7.7.3 *Results and discussion.* The gross radiation exposure rates (which include the natural radiation background) in Table 7.7 range from 6.9 to 10.2 µR/h in the immediate environs, and up to 69 µR/h on-site at the measurement locations. Radiation values at background locations 3 and 10 averaged 8.3 ± 0.5 µR/h, and all off-site values averaged 8.1 ± 0.8 (1-sigma) µR/h. Because of differences in the natural background at various locations, none of the measured exposure rates off-site could be shown to include radiation from the station.

The higher radiation exposure rates on-site are attributed primarily to direct radiation from stored radioactive waste. This explanation was supported qualitatively by the general decrease in exposure rates with distance from the boron waste storage tanks, and the lower values where buildings provided shielding. Since all measurements were taken while the reactor was operating, it is not possible to eliminate direct radiation from the reactor as a possible source. The responses of the survey meters in moving from location to location suggested that the higher exposure rates were not due to higher radiation background, radionuclides from the station deposited on the ground, or radiation from airborne radioactive effluents.

An attempt was made to evaluate the exposure rates off-site by extrapolating from the higher values measured on-site. The distance of each location from the boron waste storage tanks, at the north corner of the plant, appeared to be the critical measurement in correlating the dose and distance measurements. The values were extrapolated to off-site locations by the equation:

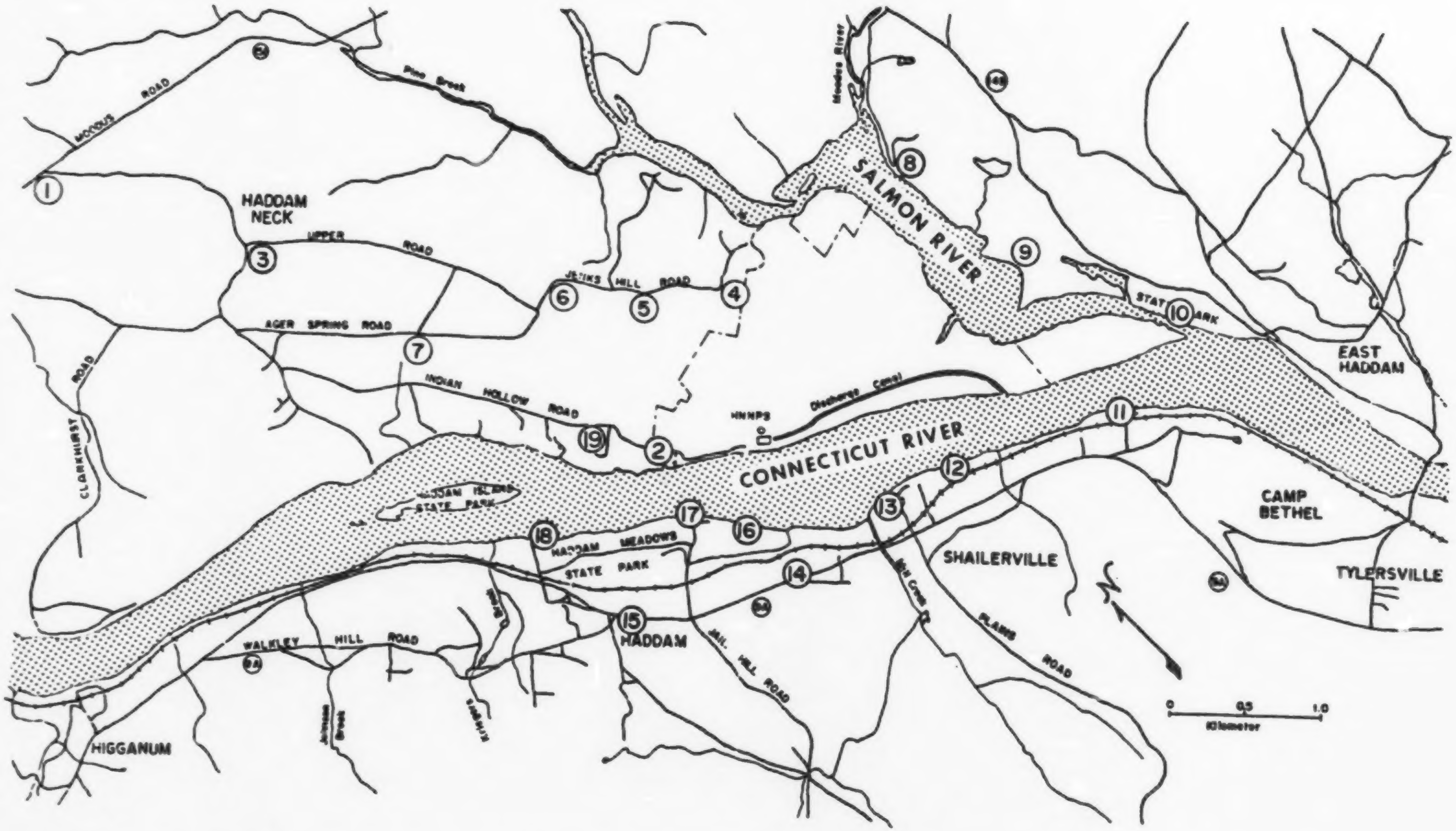
1
HADDAM NECK
MOODUS ROAD
Pine Brook
3
UPPER ROAD
AGER SPRING ROAD
7
CLARKHURST ROAD
6
JENKS HILL ROAD
5
4
INDIAN HOLLOW ROAD
19
2
HNNPS
Discharge Canal
Moodus River
8
SALMON RIVER
9
10
STATE PARK
EAST HADDAM
11
CONNECTICUT RIVER
12
13
CAMP BETHEL
TYLERSVILLE
SHAILERVILLE
PLAINS ROAD
HADDAM ISLAND STATE PARK
18
17
16
HADDAM MEADOWS STATE PARK
14
15
HADDAM
JAIL HILL ROAD
WALKLEY HILL ROAD
HIGGANUM
N
0
0.5
1.0
Kilometer

Figure 7.2 Locations of Off-Site Radiation Exposure Measurements with Survey Meters

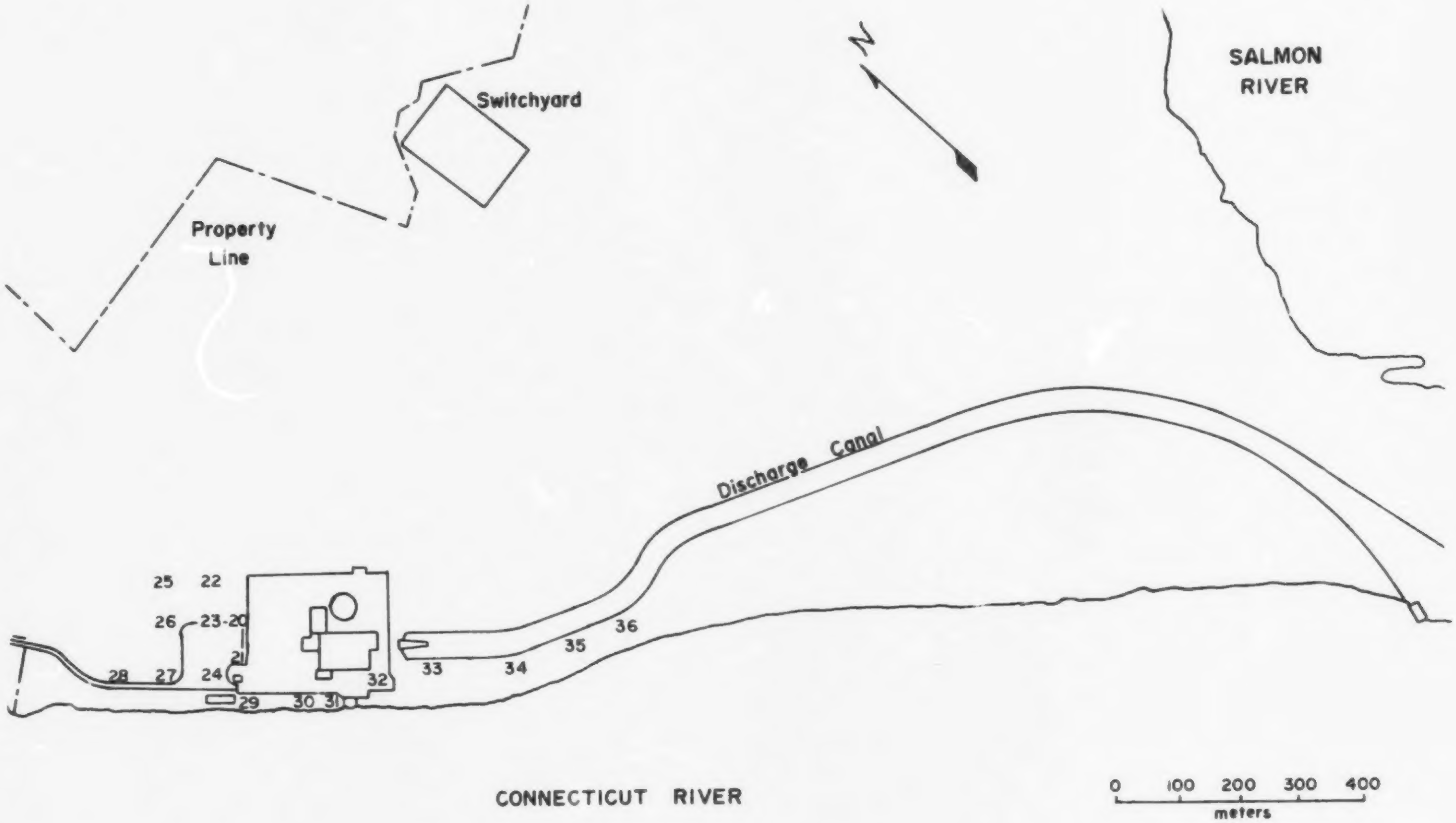
SALMON RIVER
Switchyard
Property Line
Discharge Canal
CONNECTICUT RIVER
25
22
26
23-20
21
28
27
24
29
30
31
32
33
34
35
36
0
100
200
300
400
meters

Figure 7.3 Locations of On-Site Radiation Exposure Measurements with Survey Meters

Table 7.7

External Radiation Exposure Rates Near Haddam Neck

Location No.	Location Distance[a]	Exposure rate on July 22, 1970, μR/hr	Location No.	Location Distance	Exposure rate on March 16, 1971, μR/hr
	Off-site			**On-site**	
1	5.0 km NNW	10.2 ± 0.4[b]	20	0.18 km NW	69 ± 9
2	0.5 km NW	8.1 ± 0.1	21	0.20 km WNW	38 ± 7
		9.0 ± 0.5[c]	22	0.22 km NW	34 ± 3
3	3.7 km NNW	8.7 ± 0.4	23	0.21 km NW	30 ± 4
4	1.0 km NNE	8.8 ± 0.0	24	0.24 km WNW	21 ± 2
5	1.2 km N	8.0 ± 0.1			24 ± 1[d]
6	1.6 km N	8.3 ± 0.2	25	0.28 km NW	18.3 ± 0.8
7	2.3 km NNW	8.6 ± 0.3	26	0.28 km NW	15.0 ± 0.4
8	2.0 km ENE	8.2 ± 0.1	27	0.31 km WNW	12.1 ± 0.2
9	2.1 km E	7.9 ± 0.4	28	0.38 km WNW	10.1 ± 0.5
10	3.0 km ESE	7.8 ± 0.5	29	0.20 km W	26 ± 1
11	2.4 km SE	7.1 ± 0.2	30	0.14 km WSW	32 ± 1
12	1.3 km SE	7.2 ± 0.3	31	0.13 km SW	24 ± 1
13	1.0 km SSE	6.9 ± 0.2	32	0.12 km SSW	41 ± 1
14	1.0 km SSW	8.6 ± 0.2	33	0.18 km S	55 ± 3
15	1.5 km W	8.3 ± 0.2	34	0.31 km SSE	31 ± 2
16	0.7 km WSW	7.9 ± 0.3	35	0.39 km SSE	14.8 ± 0.5
17	0.8 km W	7.5 ± 0.1	36	0.47 km SE	10.1 ± 0.3
18	1.5 km WNW	7.0 ± 0.3			
19	0.6 km NW	8.3 ± 0.2[c]			

[a] Distance from center of reactor containment

[b] Exposure rates are averages of 2 to 8 measurements; ± values are 1/2 of the range for 2 measurements or 2σ values for more than 2 measurements.

[c] Measured on March 16, 1971

[d] Measured on July 22, 1970

$$R = 1.4D^{-2}\exp(-4D) \qquad (7.2)$$

where R is the net radiation exposure rate (background subtracted) in μR/h and D is the distance in kilometers from the boron waste storage tanks. The constant of 1.4 in this equation was obtained from the net exposure rates at locations 21–28 by a least-squares evaluation of the data. The exponential constant of 4 accounts for the attenuation of the gamma radiation in air.

The extrapolated values are as follows:

No.	Location	Distance from boron waste waste storage tanks, m	Extrapolated exposure rate, μR/h
2	(NW perimeter)	460	1.1
19	(NW nearest residence)	640	0.3
14	(Haddam)	1000	0.1
	River in front of plant (100 m from shore)	240	9.3
15	(Haddam)	1500	0.03
16	(Haddam)	700	0.2
17	(Haddam)	800	0.2

The extrapolated radiation rates were 1 μR/h or less at the site boundary (except on the Connecticut River) and 0.2–0.3 μR/h at the nearest residences, both NW of the site and across the river in Haddam. These values are consistent with the measured gross values, but are uncertain because the extrapolation is overly simple in assuming the same relation in all directions and ignoring radiation shielding and scattering. The exposure rate on the river, 100 meters offshore from the station, was estimated to be approximately 9 μR/h above background.

The station's contractor for environmental surveillance has reported(2) gross gamma radiation exposures averaging from 22 to 26 μR/h at the station and eight neighboring towns. These TLD values seem questionable in view of the lower values measured in this study and the generally encountered lower levels of background radiation.

It would be of interest to obtain additional measurements of external radiation exposure in the environs of the station to check the presented data with regard to instrument calibration, background subtraction, and possible correlation with station operation, including the radiation levels of the stored radioactive waste. Long-term measurements of continuous gamma-ray sources to detect levels 0.3 μR/h above background do not appear feasible even with sensitive detectors, since it would be difficult to distinguish between radiation from the station and the natural background.

7.7.4 *Estimated external radiation exposure to persons in the environs.* The instantaneous exposure rate from the station of 0.3 μR/h at the nearest location, computed in Section 7.7.3, equals 2.6 mR/yr. The estimated exposure to individuals in Haddam would then be 2 mR/yr or less. These values are subject to considerable uncertainties in measurement and calculation, and will be reduced by shielding by house walls and time spent by persons at less exposed locations. In comparison, the natural radiation background was approximately 70 mR/yr and its variation was much greater than the exposure attributed to the station. Persons in boats on the river less than 1 km from the plant would be subject to higher exposure rates, but for shorter periods of time. In 100 hours per year at a location 100 meters from the nearest shore line in front of the station, the annual dose would be 1 mR. The set of measurements suggests that the radiation exposure from radioactive wastes stored at the station was essentially zero at distances of 2 km and more.

The exposure due to direct radiation from the boron waste storage tanks was estimated in the Environmental Statement to be 5.5 mrem/yr at the site boundary and about 0.9 mrem/yr at the nearest residence.(4) This was calculated from an assumed radionuclide inventory in the tanks and the shielding thickness of the tank walls. A similar calculation for direct exposure from the waste gas surge sphere resulted in an estimate of 6 mrem/yr at the nearest boundary. Due to the different locations of the storage tanks and the waste gas surge sphere, no off-site location would be exposed to the sum of these maxima.

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8. SUMMARY AND CONCLUSIONS

8.1 Radionuclides in Effluents from the Haddam Neck Station

Radionuclides were discharged by numerous pathways in small amounts relative to effluent limits. The largest constituents among radioactive effluents were ^{3}H, mostly in liquid waste, and ^{133}Xe, mostly in airborne waste. These observations appear to be generally applicable to large PWR nuclear power stations, except that less ^{3}H is discharged when the fuel is clad in Zircaloy instead of stainless steel. Lesser discharges of many radionuclides, including ^{133}Xe, have been predicted when additional waste treatment is applied to meet 'as low as practicable' criteria for design objectives.

Results of the effluent measurements in this study are summarized below, based on the information in Sections 3 and 4. For simplicity, they are given as annual releases. Because these values were obtained by occasional sampling, they should be considered only indications of the magnitude of radionuclide discharges. Exact values must be derived from frequent or continuous measurements at the many waste streams or discharge locations. The totals (see Sections 3.3.13 and 4.3.3) are comparable to the annual discharges reported by the station operator and the discharge estimates presented in the Environmental Statement for the station; any differences in individual values are discussed in Sections 3 and 4. The largest discrepancy concerns calculations based on a 'model' plant: the amounts of discharged gaseous radionuclides are similar to those predicted for 0.25 percent of fuel elements releasing radionuclides to the coolant, although this value was only 0.02 percent at the Haddam Neck station.

The estimated amounts of radionuclides in airborne effluents during the second half of 1970 and the first half of 1971 are as follows:

Radionuclides in airborne effluents, Ci/yr

Radionuclide	(1) Waste gas surge sphere	(2) Vapor container air	(3) Primary auxiliary bldg air	(4) Steam jet air ejector	(5) Fuel bldg air	(6) Primary vent stack	(7) Turbine bldg air
^{3}H	0.007	0.18	4.3	0.081	2.3	12.	150.
^{14}C	0.032	0.18	$<2.$	0.051	0.3	$<3.$	$<10.$
^{85m}Kr	na	na	na	6.6	na	na	na
^{85}Kr	29.	77.	9.1	12.	0.8	26.	43.
^{87}Kr	na	na	na	10.	na	na	na
^{88}Kr	na	na	na	17.	na	na	na
^{133m}Xe	0.4	0.3	na	18.	na	na	na
^{133}Xe	130.	120.	650.	1,100.	$<12.$	1,500.	$<400.$
^{135}Xe	0.50	na	na	58.	na	na	na
^{131}I	0.0007	na	na	na	na	0.013	na
Long-lived particulate	0.0003	na	na	na	na	0.003	na

note: na = not analyzed

The values in data column 6 include the same wastes as those in columns 3, 4, and 5, and also discharges from the blowdown flash tank, which could not be measured separately. The ^{3}H and ^{14}C values are for all forms of the radionuclides, but distinctions between tritiated water and gases, and between ^{14}C in CO_2 and other gases, are made in Section 3 for several of the waste streams. A small amount of ^{41}Ar was also observed in one of the gaseous wastes. The presence of ^{83m}Kr, ^{89}Kr, ^{131m}Xe, ^{135m}Xe, ^{137}Xe, and ^{138}Xe in curie amounts was inferred, although these radionuclides could not be measured. Short-lived progeny of noble gases, such as ^{88}Rb, and relatively short-lived iodine isotopes, such as ^{133}I and ^{135}I, were also indicated to be present. Some of the listed values in ventilating air were noted to be uncertain because too few samples were collected. A few totals are incomplete because a contributing waste stream could not be sampled.

The tabulation suggests that the usual program of radioactive gas measurements at the air ejector and in waste tanks can account for a large fraction of the discharges. Discharged ventilating air, however, also carries radioactivity by a variety of pathways. The radionuclide compositions of the various effluents differ widely.

The estimated amounts of radionuclides in liquid effluents during the same period are as follows:

Radionuclides in liquid effluents, Ci/yr

Radionuclide	Reactor system waste	Secondary system waste blowdown	Secondary system waste leakage
^{3}H	8×10^{3}	6×10^{1}	2×10^{2}
^{14}C	3×10^{-2}	6×10^{-4}	
^{24}Na	ND	2×10^{-2}	
^{32}P	5×10^{-4}	7×10^{-3}	
^{51}Cr	3×10^{-3}	2×10^{-3}	
^{54}Mn	1×10^{-1}	2×10^{-3}	
^{55}Fe	4×10^{-1}	4×10^{-2}	
^{59}Fe	8×10^{-3}	(6×10^{-4})	
^{57}Co	6×10^{-3}	(2×10^{-4})	
^{58}Co	4×10^{-1}	2×10^{-2}	
^{60}Co	2×10^{-1}	1×10^{-3}	
^{63}Ni	6×10^{-3}	(2×10^{-3})	
^{89}Sr	4×10^{-4}	1×10^{-4}	
^{95}Zr	5×10^{-4}	2×10^{-4}	
^{95}Nb	3×10^{-3}	(2×10^{-4})	
^{99}Mo	$<1 \times 10^{-3}$	3×10^{-3}	
^{110m}Ag	3×10^{-3}	ND	
^{131}I	4	2	
^{133}I	1×10^{-1}	2	
^{135}I	ND	8×10^{-1}	
^{133}Xe	2×10^{1}	ND	
^{134}Cs	1×10^{-1}	8×10^{-2}	
^{136}Cs	$<1 \times 10^{-3}$	2×10^{-2}	
^{137}Cs	2×10^{-1}	8×10^{-2}	

note: ND-not detected

Tritium was assumed to be the only radionuclide in water leaking from the secondary system, although no confirmatory measurements were made. The values in parentheses were inferred as described in Section 4.3.3. Unlisted radionuclides, such as ^{90}Sr, were not detected in any effluent samples; less-than values for some are given in Section 4.3.3.

The bulk of the effluent radioactivity was discharged from the reactor waste system. A few radionuclides, however, were in higher amounts in secondary system blowdown, which is discharged without storage or treatment.

8.2 Radionuclides in the Environment at the Haddam Neck Station

Radionuclides at low concentrations from the station were found in various media sampled in the aquatic environment:

(1) The radionuclides ^{3}H and ^{131}I were in water at concentrations of approximately 10 pCi/ml and 1 pCi/liter, respectively. The radionuclides ^{58}Co and ^{134}Cs were at concentrations between 0.1 and 1 pCi/liter. Samples with these contents were obtained in the coolant water discharge canal and within a few kilometers of its mouth in the Connecticut River (see Sections 4.3.4 and 5.2).

(2) Numerous radionuclides were in algae, plankton, and aquatic plants collected at the mouth of the canal and nearby in the Connecticut River. Iron-55 had the highest concentration in these samples, at 41 pCi/g wet weight (see Section 5.4).

(3) Fish caught in the canal contained ^{3}H, ^{14}C, ^{131}I, ^{134}Cs, and ^{137}Cs. The highest concentration, of ^{3}H, was 2.9 pCi/g wet weight of tissue, compared to a background value of 0.6 pCi/g. One sample of shad, which swim up the Connecticut River for a brief period to spawn, showed a similar increase in ^{3}H concentration, but contained no other radionuclides attributed to the station (see Section 5.5). Oysters and clams from the mouth of the Connecticut River had no elevated levels of radionuclides (see Section 5.6). No shellfish were found in or near the coolant canal, although their presence had been reported.

(4) Sediment from some locations that accumulate silt along the east bank of the Connecticut River at and just above and below the mouth of the canal contained ^{54}Mn, ^{57}Co, ^{58}Co, ^{60}Co, ^{134}Cs, and ^{137}Cs attributed to station effluents. The presence of ^{58}Co was usually most apparent; its highest concentration was 13.5 pCi/g (see Section 5.7).

Radioactive effluents and direct radiation from the station could not be readily detected in the terrestrial environment. No radionuclides attributed to the station were found in well water from just beyond the station boundary, in vegetables from nearby gardens, in milk from nearby dairy farms, or in the meat of deer killed near the station (see Sections 7.2, 7.3, 7.4, and 7.6). The following special measurements showed radionuclides or radiations from the station in the environment:

(1) The thyroid of one dairy cow that had grazed on a hill 3.3 km distant from the station contained 21 pCi of ^{131}I (0.4 pCi/g thyroid) at the time when the ^{131}I discharge from the stack was approximately 10^{-3} µCi/s (see Section 7.5). A second thyroid, from a cow that had grazed at the same distance but at a lower elevation and during lesser ^{131}I discharges, contained no detectable ^{131}I (< 2 pCi).

(2) While gas from the surge sphere was discharged for this purpose, the presence of ^{133}Xe in ground-level air was observed with a large, thin NaI(Tl) detector (FIDLER) with pulse height discrimination to count the characteristic 81-keV gamma rays. Measurements of ^{133}Xe at a stack discharge rate of approximately 2,000 µCi/s were possible on site. By collecting 0.5 m^3 of air, ^{133}Xe and ^{85}Kr were detected in ground level air at concentrations of 2×10^{-2} µCi/m^3 and 4×10^{-3} µCi/m^3, respectively. Indications of ^{133}Xe during releases from the waste gas surge sphere could occasionally be obtained off-site with the FIDLER survey instrument. Such measurements were not sufficiently sensitive to detect the much lower amounts of gaseous radioactivity discharged continuously from the stack (see Section 6).

(3) Measurements with survey meters beyond the station boundary showed no observable increase over the background radiation exposure of approximately 8 µR/h (70 mR/yr). Extrapolation of elevated radiation exposures within the boundary suggested that the highest exposure rate at nearby habitations was 0.3 µR/h (2.6 mR/yr) due to direct radiation from the station. The exposure rate was estimated to be lower at the nearest population center, but higher on the Connecticut River, where persons would be exposed only briefly (see Section 7.7). The source of the radiation is believed to be stored radioactive waste.

On the basis of the observed effluent and on-site measurements, the highest population radiation doses were computed to be from consuming fish caught in and near the coolant-water discharge canal and from external radiation due to stored wastes and gaseous discharges:

(1) fish consumption (Section 5.5.5) may have resulted in 3 mrem/yr to bone, 6 mrem/yr to thyroid, 0.4 mrem/yr to GI tract, and 0.3 mrem/yr to the total body;

(2) direct radiation (Section 7.7.4) may have resulted in 3 mrem/yr to the total body;

(3) airborne discharges (Section 3.3.14) may have resulted in 0.5 mrem/yr to the total body.

The computations—particularly those of the dose from fish consumption—utilized several assumptions that require checking. The external radiation doses would be lower if adjusted for shielding and occupancy factor.

8.3 Monitoring Procedures

The following procedures were demonstrated in this and previous studies for monitoring effluents and environments of PWR stations:

(1) measurement of effluent radionuclides other than the long-lived ones readily detected by a gamma-ray spectrometer; of particular interest, in addition to usually measured ^{3}H, ^{89}Sr, and ^{90}Sr, are ^{14}C, ^{32}P, and ^{55}Fe;

(2) measurement of ^{3}H and ^{14}C in their various gaseous species. Other recent studies suggest the inclusion of species measurements for radioiodine in air and radiocobalt in the aquatic environment;

(3) surveillance of sediment with submersible gamma-ray detectors to indicate "hot spots" for detailed sampling and analysis;

(4) use of concentration devices to collect ionic and insoluble radionuclides from water for measurements at concentrations of 10^{-10} µCi/ml;

(5) use of bovine thyroids to detect ^{131}I at very low concentrations (equivalent to 0.02 pCi/liter milk) in the terrestrial environment;

(6) use of specialized survey instruments for detecting low levels of ^{133}Xe (400 pCi/m^3) in ground-level air;

(7) collection of fish that are under conditions of relatively restricted mobility in the environment to study their uptake of radionuclides.

8.4 Recommendations for Environmental Surveillance

The observations in this study support previously presented recommendations that assessment of population radiation exposures from routine facility operation be based on measuring the radioactive effluent and radiation flux at the station. Environmental radionuclide and radiation levels attributable to station operation were generally too variable, obscured by the radiation background, or near instrumental detection limits to be measured precisely for evaluating exposure. Measurements at the source must include all significant pathways and radionuclides during the entire period of operation; critical radionuclides can be missed by monitoring only the obvious effluents and the easily measured radionuclides, or by ignoring the effects of changes in the operating cycle. Detailed studies of in-plant radionuclide pathways are needed for selecting an optimum program for sampling and analysis.

Environmental measurements were found to be useful for supporting and confirming the population radiation exposures computed from on-site monitoring, and for providing these computations with numerical factors applicable to the site. Such measurements, if performed reliably, can also be reassuring in demonstrating that no unexpected radioactivity is in the environment. For a station and site such as Haddam Neck, the following measurements provide useful information:

(1) confirmation of critical pathways
  a) measure critical radionuclides in fish caught in and near the coolant canal
  b) measure external radiation exposure rates on site and the decrease of the exposure rate as function of distance to off-site locations

(2) determination of numerical factors for computing doses
  a) compute X/Q values by measuring ^{133}Xe concentration in ground-level air relative to the release rate at the station
  b) observe long-term accumulation of radionuclides in aquatic environment and possible transfer to the food chain

(3) assurance that no significant exposure exists from unforeseen sources or occasional operational occurrences
  a) measure radiation exposure at nearby habitations
  b) measure radionuclides in milk, food, and drinking water obtained in immediate vicinity of station.

This program by EPA and cooperating groups will be concluded with a similar study at a commercially operated BWR nuclear power station. Generic radiological surveillance studies at other facilities in the nuclear fuel cycle are under consideration.

Appendix A
Acknowledgments

This report presents the work of the staff of the Radiochemistry and Nuclear Engineering Facility, USEPA, consisting of the following:

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Appendix B.1

Haddam Neck Average Monthly Power and Reactor Coolant Chemistry Statistics from Monthly Operating Reports

		Average Power		Main Coolant*						Primary Plant	
Year	Month	MWt	MWe (gross)	Boron, ppm	Tritium, μCi/ml	^{131}I, 10^{-3} μCi/ml	pH	Crud, ppm	Gross Activity, 10^{-1} μCi/ml	Leak Rate, l/min	Remarks
1967	Criticality achieved 1:04 am July 24, 1967. No significant operation during August and September.										
	Oct.	240	76	< 1657	0.05	N.S.	5.6	0.39	0.5		
	Nov.	627	209	< 1633	0.30	0.39	5.5	0.18	1.2		
	Dec.	1413	471	1630	0.77	0.52	5.5	0.07	1.3		
1968	Jan.	969	322	~ 1650	0.97	2.47	5.6	0.17	1.3		
	Feb.	1448	482	~ 1650	0.14	0.26	6.0	0.08	1.4		
	Mar.	81	27	1850	0.70	0.29	5.7	0.06	0.5		
	Apr.	512	170	1600	0.36	0.47	5.6	0.20	0.5		
	May	1436	479	< 1530	1.19	0.15	6.3	0.06	2.0		
	June	1188	396	1490	1.30	0.18	6.2	0.07	1.8		
	July	1227	409	1415	1.46	0.09	6.3	0.04	1.8		
	Aug.	954	318	1375	1.58	3.68	6.3	< 0.01	2.2		
	Sept.	1193	398	1295	1.29	0.39	6.3	~ 0.01	1.5		
	Oct.	1466	492	1211	2.39	0.51	6.3	~ 0.01	2.3		
	Nov.	1249	420	1144	3.45	0.13	6.4	0.06	1.7		
	Dec.	1239	420	1080	3.00	0.27	6.5	0.34	1.5		
1969	Jan.	1420	482	993	2.87	0.23	6.6	< 0.02	1.6	0.4	
	Feb.	1402	476	918	3.81	0.10	6.6	< 0.10	1.5	0.4	
	Mar.	1577	532	806	3.67	0.13	6.7	< 0.01	1.7		
	Apr.	648	218	710	4.54	0.18	6.7	< 0.01	1.6		1.
	May	771	258	728	1.57	10.96	6.6	0.08	1.0		1.
	June	1148	376	678	1.23	0.29	6.7	0.03	1.1	0.6	
	July	1274	414	610	1.59	0.26	6.8	0.13	1.0	0.4	
	Aug.	1309	427	533	2.03	0.26	6.9	< 0.01	1.0	0.6	
	Sept.	1476	482	425	1.43	0.40	7.1	0.03	1.1	0.6	
	Oct.	1595	530	312	2.61	1.31	6.9	0.20	1.4	0.5	2.
	Nov.	1436	478	234	1.63	1.54	7.2	0.04	1.5	0.5	
	Dec.	1747	584	155	2.16	9.10	6.7	0.02	1.5	0.6	
1970	Jan.	1799	596	51	2.11	3.78	6.5	0.01	1.5	0.7	3.
	Feb.	1654	545	0.1	1.71	2.98	7.7	0.04	1.6	2.1	4.
	Mar.	1337	433	< 1	0.87	1.48	8.5	< 1.00	1.0	0.8	
	Apr.	740	234	< 1	0.96	13.4	8.1	< 0.11	1.1	0.8	5.
	May	0	0	2300	0.10	N.S.	N.S.	N.S.	1.5		5.
	June	79	22	776**	0.57	0.14	5.4	N.S.	0.1		5.
	July	1724	551	574	5.43	4.65	6.2	0.03	2.0	1.2	
	Aug.	1543	500	516	5.38	5.86	6.9	< 0.02	2.0	1.3	
	Sept	1483	481	447	1.77	13.8	6.3	0.03	2.2	3.9	
	Oct.	1724	568	371	2.94	56.0	6.3	< 0.02	2.6	1.9	
	Nov.	1825	586	287	3.68	27.9	6.8	< 0.02	2.6	1.8	
	Dec.	1739	576	207	3.39	85.	6.9	< 0.01	3.5	1.8	

Appendix B.1 (cont'd)

Haddam Neck Average Monthly Power and Reactor Coolant Chemistry Statistics from Monthly Operating Reports

		Average Power		Main Coolant*						Primary Plant	
Year	Month	MWt	MWe (gross)	Boron, ppm	Tritium, µCi/ml	^{131}I, 10^{-3} µCi/ml	pH	Crud, ppm	Gross Activity, 10^{-1} µCi/ml	Leak Rate, l/min	Remarks
1971	Jan.	1792	596	102	3.29	34.	7.1	0.01	4.1	1.8	
	Feb.	1763	588	15	3.07	45.3	8.5	<0.01	5.0	1.6	3.
	Mar.	1720	571	0.6	4.52	29.5	10.0	<0.01	5.0	1.9	6.
	Apr.	739	275	0.9	3.41	19.	9.9	<0.01	3.7	2.0	7.
	May	231	78	1243**	0.14	25.4	6.3	<0.01	1.7	1.5	7.
	June	1582	517	930	1.9	17.4	6.7	0.01	4.5	1.5	
	July	1713	549	875	3.67	15.8	6.6	0.11	4.9	1.4	
	Aug.	1728	555	798	5.17	16.5	6.6	<0.01	6.4	0.9	
	Sept.	1760	574	709	5.63	15.6	6.8	0.03	6.7	0.7	
	Oct.	1692	556	620	4.14	21.3	6.8	0.01	6.1	0.7	
	Nov.	1813	601	511	7.00	17.5	6.8	0.01	5.3	0.6	
	Dec.	1733	575	446	6.23	17.6	6.8	0.01	4.4	0.9	

* Average of reported values

** While reactor was operating

N.S. – No Sample

Remarks:

1. Shutdown for turbine modification 4/11/69 to 5/11/69.
2. Bleed and feed operation toward end of month to reduce fission product gases in coolant.
3. Bleed and feed operation to reduce boron in main coolant.
4. Coastdown began 2/7/70.
5. Shutdown for refueling 4/17/70 to 6/24/70.
6. Coastdown began 3/5/71.
7. Shutdown for refueling 4/16/71 to 5/21/71.

Appendix B.2

Haddam Neck Radioactive Waste Discharges from Monthly Operating Reports

		Liquid			Gaseous		
Year	Month	Volume, liters	Gross β-γ, mCi	Tritium, Ci	Volume, m^3	Gross β-γ, Ci	Tritium, mCi
1967	Oct.		20	2		<0.1	
	Nov.		71	123			
	Dec.		125	95		none	
1968	Jan.		131	215		0.6	
	Feb.		47	60		none	
	Mar.		356	337		<0.1	
	Apr.		1,619	119		<0.1	
	May		162	85		none	
	June		31	137		<0.1	
	July	2.1×10^5	34	27	1.4×10^3	2.6	9
	Aug.	3.5×10^5	142	170	8.5×10^1	<0.1	
	Sept.	3.6×10^5	1,102	159	2.1×10^4	0.3	
	Oct.	1.4×10^5	233	66		none	9,060
	Nov.	3.1×10^5	13	166		none	
	Dec.	3.2×10^5	99	195		none	
1968	Total		3,969	1,735		3.7	9,069
1969	Jan.	2.5×10^5	28	269		none	
	Feb.	2.0×10^5	16	160		none	
	Mar.	1.9×10^5	4	261	2.1×10^3	<0.1	1
	Apr.	2.9×10^5	51	624	6.1×10^5	<0.1	2,500
	May	4.0×10^5	10,740	669	4.7×10^2	1.0	1
	June	2.6×10^5	1,320	138	4.7×10^2	0.9	3
	July	3.7×10^5	105	181	6.6×10^2	2.9	5
	Aug.	1.6×10^5	33	156	5.7×10^1	0.3	30
	Sept.	4.5×10^5	5	360	9.2×10^1	0.1	0
	Oct.	6.3×10^5	3	943	1.3×10^3	129.9	5
	Nov.	3.9×10^5	12	560	6.3×10^2	44.7	3
	Dec.	6.4×10^5	525	730	9.3×10^1	5.1	0
1969	Total	4.3×10^6	12,842	5,051		185.0	2,548
1969	Total Reported*	4.3×10^6	12,170	5,163		185.2	2,520

Appendix B.2 (cont'd)

Haddam Neck Radioactive Waste Discharges from Monthly Operating Reports

		Liquid			Gaseous		
Year	Month	Volume, liters	Gross β-γ, mCi	Tritium, Ci	Volume, m^3	Gross β-γ, Ci	Tritium, mCi
1970	Jan.	6.9×10^5	638	1,078	1.1×10^3	18.7	0
	Feb.	2.3×10^6	1,410	826	9.9×10^3	70.4	n.r.††
	Mar.	2.9×10^6	474	434	2.8×10^3	1.4	0
	Apr.	1.0×10^6	4,640	441	7.9×10^5	326.3	n.r.
	May	6.7×10^5	1,040	112	2.4×10^5	16.9	n.r.
	June	1.2×10^6	3,187	56	5.1×10^5	2.7	81
	July	4.6×10^5	573	138	1.7×10^4	8.7	n.r.
	Aug.	4.9×10^5	2,749	828	1.6×10^4	12.8	n.r.
	Sept.	2.9×10^6	10,863	1,494	1.6×10^4	234.2	n.r.
	Oct.	2.5×10^6	1,150	441	1.9×10^4	46.4	n.r.
	Nov.	2.7×10^6	2,347	751	1.9×10^4	56.3	n.r.
	Dec.	2.5×10^6	1,867	753	2.0×10^4	80.4	n.r.
1970	Total	2.0×10^7	30,938	7,354	1.7×10^6	875.2	81
1970	Total Reported**		22,085	7,377		701.5	123
1971	Jan.	1.6×10^6	2,016	879	1.9×10^4	93.0	0
	Feb.	2.5×10^6	7,509	993	1.9×10^4	137.8	n.r.
	Mar.	2.6×10^6	1,380	550	1.5×10^4	133.1	n.r.
	Apr.	2.3×10^6	18,854	1,576	6.3×10^6	2,760.	605
	May	2.9×10^6	1,765	291	2.9×10^6	19.	271
	June	2.6×10^6	1,436	189	1.2×10^4	11.6	n.r.
	July	2.2×10^6	294	134	1.2×10^4	13.2	n.r.
	Aug.	2.5×10^6	142	221	5.3×10^4	15.7	n.r.
	Sept.	2.3×10^6	409	88	1.4×10^7	16.0	n.r.
	Oct.	2.5×10^6	304	487	1.7×10^4	31.9	n.r.
	Nov.	2.4×10^6	1,334	140	1.2×10^4	23.6	n.r.
	Dec.	2.4×10^6	597	280	1.3×10^4	34.7	n.r.
1971	Total	2.9×10^7	36,040	5,830	2.3×10^7	3,289.6	876
1971	Total Reported†	2.9×10^7	35,896	5,832	2.3×10^7	3,251.0	876

* Connecticut Yankee Atomic Power Company, Operation Report No. 70–1, Jan. 1970.

**Connecticut Yankee Atomic Power Company, Operation Report No. 70–12, Dec. 1970.

† Connecticut Yankee Atomic Power Company, Operation Reports 71–6, June 1971, and 71–12, Dec. 1971.

††n.r. – none reported

Note:

Total liquid gross β-γ discharged, excluding ^{133}Xe and ^{135}Xe was:
1970 6.67 Ci
1971 5.9 Ci

Appendix B.3*

Haddam Neck Radionuclide Discharges

	1970		1971	
	January-June	July December	January-June	July-December
	Liquid			
Total volume (liters)	8.7×10^6	1.5×10^7	1.5×10^7	1.4×10^7
Reactor system volume –				
borated (liters)	n.r.**	$14. \times 10^5$	$15. \times 10^5$†	3.1×10^5†
aerated (liters)	n.r.	$19. \times 10^5$	20.9×10^5†	9.6×10^5†
Not identified (mCi)	702	883	21	32
^{3}H (Ci)	2,970	4,406	4,478	1,352
^{54}Mn (mCi)	n.r.	97	3	404
^{58}Co (mCi)	3,444	494	764	37
^{60}Co (mCi)	6	7	79	734
^{131}I (mCi)	72	687	1,750	373
^{133}I (mCi)	n.r.	178	716	326
^{133}Xe (mCi)	8,269	7,103	28,800	1,080
^{135}Xe (mCi)	7	33	110	48
^{137}Cs (mCi)	20	84	575	43
Total % Allowable (based on isotopic analysis)	4.04	3.99	2.44	0.62
Total Available Dilution (liters)	2.7×10^{11}	3.8×10^{11}	3.5×10^{11}	3.9×10^{11}
	Gaseous			
Volume (m^3)	n.r.	n.r.	9.8×10^6	1.4×10^7
^{3}H (mCi)	120††	n.r.	876	n.r.
^{41}Ar (Ci)	n.r.	3††	n.r.	n.r.
^{85}Kr (Ci)	7	3	18	n.r.
^{88}Rb (mCi)	n.r.	n.r.	41	159
^{131}I (mCi)	n.r.	n.r.	12	0.03
^{133}I (mCi)	n.r.	n.r.	0.9	0.05
^{133}Xe (Ci)	188	415	2,893	109
^{135}Xe (Ci)	7	21	204	16
^{135m}Xe (Ci)	n.r.	n.r.	0.7	10
Particulates (mCi)	1,410††	n.r.	8	10
Total (Ci)	219††	483††	3,116	135
Total % Allowable (Based on isotopic analysis)	0.25	2.55	0.18	
Total Available Dilution (m^3)	n.r.	n.r.	5.1×10^8	5.3×10^8

* Connecticut Yankee Atomic Power Company, Operation Reports 70-12, 71-6, and 71-12, Dec. 1970, June 1971, and Dec. 1971.

**n.r. – none reported

† Connecticut Yankee Atomic Power Company, Operation Reports 71-1 to 71-12, Jan-Dec 1971.

††Expressed as ^{133}Xe equivalent

Appendix B.4*

Sources of Waste at Haddam Neck—July-December, 1970

Radionuclide or Volume	July	Aug.	Sept.	Oct.	Nov.	Dec.
Liquid waste from reactor system test tanks						
Gross β-γ, mCi (unidentified)	435	5	208	109	n.r.	n.r.
^{3}H, Ci	111	826	1,485	429	730	720
^{54}Mn, mCi	n.r.**	n.r.	n.r.	56	28	13
^{58}Co, mCi	8	80	60	n.r.	319	27
^{60}Co, mCi	n.r.	n.r.	n.r.	n.r.	7	n.r.
^{131}I, mCi	n.r.	n.r.	479	31	6	n.r.
^{133}I, mCi	n.r.	n.r.	37	n.r.	n.r.	n.r.
^{133}Xe, mCi	86	2,653	10	925	1,760	1,670
^{135}Xe, mCi	7	2	4	2	17	2
^{137}Cs, mCi	n.r.	n.r.	n.r.	n.r.	74	10
Volume, liter	4.6×10^{5}	4.9×10^{5}	9.0×10^{5}	3.8×10^{5}	7.3×10^{5}	3.5×10^{5}
Liquid waste from secondary system: leakage and blowdown						
Gross β-γ,mCi (unidentified)	34	9	55	21	2	1
^{3}H, Ci	27	3	9	12	23	32
^{131}I, mCi	n.r.	n.r.	n.r.	21	66	84
^{133}I, mCi	n.r.	n.r.	n.r.	12	69	61
Volume, liter	n.r.	n.r.	2.0×10^{6}	2.1×10^{6}	2.0×10^{6}	2.1×10^{6}
Gaseous waste from air ejector						
^{133}Xe, Ci	5.2	12.1	44.6	42.5	49.8	71.6
^{135}Xe, Ci	3.1	0.4	2.6	4.0	6.4	4.1
^{41}Ar, Ci	0.4	n.r.	n.r.	n.r.	n.r.	n.r.
Volume, m^{3}	1.7×10^{4}	1.6×10^{4}	1.6×10^{4}	1.9×10^{4}	1.9×10^{4}	2.0×10^{4}
Gases from waste gas sphere						
^{133}Xe, Ci		3.1×10^{-1}	1.8×10^{2}		5.2×10^{-2}	3.7×10^{0}
^{135}Xe, Ci		n.r.	2.4×10^{-1}		1.0×10^{-4}	n.r.
^{85}Kr, Ci		n.r.	1.8×10^{0}		n.r.	9.8×10^{-1}
Volume, m^{3}	none	8.5×10^{-1}	1.3×10^{2}	none	1.4×10^{-1}	9.1×10^{0}

*Connecticut Yankee Atomic Power Company, Operation Reports 70–7 through 70–12, July-December 1970.

**n.r. – not reported

Appendix C.1

Calculated Generation Rate of Fission Products in Fuel at 1825 MWt Power(a)

Fission product	Yield, y(b)	Decay constant λ, s^{-1}	Generation rate μCi/s	Accumulation in 500 days, μCi
^{3}H	9.5×10^{-5}(c)	1.78×10^{-9}	2.1×10^{2}	8.6×10^{9}
^{83m}Kr	5.8×10^{-3}	1.04×10^{-4}	7.3×10^{8}	7.1×10^{13}
^{85m}Kr	1.3×10^{-2}	4.30×10^{-5}	6.8×10^{8}	1.6×10^{13}
^{85}Kr	2.9×10^{-3}	2.05×10^{-9}	7.2×10^{3}	2.9×10^{11}
^{87}Kr	2.4×10^{-2}	1.51×10^{-4}	4.5×10^{9}	3.0×10^{13}
^{88}Kr	3.5×10^{-2}	6.88×10^{-5}	3.0×10^{9}	4.3×10^{13}
^{89}Kr	4.2×10^{-2}	3.66×10^{-3}	1.9×10^{11}	5.2×10^{13}
^{89}Sr	4.5×10^{-2}	1.57×10^{-7}	8.6×10^{6}	5.5×10^{13}
^{90}Sr	5.5×10^{-2}	7.82×10^{-10}	5.2×10^{4}	2.2×10^{12}
^{95}Zr	6.4×10^{-2}	1.23×10^{-7}	9.6×10^{6}	7.8×10^{13}
^{95}Nb	6.4×10^{-2}	2.29×10^{-7}	2.1×10^{7}(d)	1.7×10^{14}
^{99}Mo	5.7×10^{-2}	2.90×10^{-6}	2.0×10^{8}	7.0×10^{13}
^{103}Ru	3.3×10^{-2}	2.03×10^{-7}	8.2×10^{6}	4.0×10^{13}
^{124}Sb	3.0×10^{-7}	1.33×10^{-7}	4.9×10^{1}	3.7×10^{8}
^{131}I	3.2×10^{-2}	9.96×10^{-7}	3.9×10^{7}	3.9×10^{13}
^{133}I	6.5×10^{-2}	9.21×10^{-6}	7.3×10^{8}	8.0×10^{13}
^{135}I	6.0×10^{-2}	2.87×10^{-5}	2.1×10^{9}	7.3×10^{13}
^{131m}Xe	4.5×10^{-4}	6.74×10^{-7}	3.7×10^{5}	5.6×10^{11}
^{133m}Xe	1.9×10^{-3}	3.57×10^{-6}	8.1×10^{6}	2.3×10^{12}
^{133}Xe	6.5×10^{-2}	1.52×10^{-6}	1.2×10^{8}	7.9×10^{13}
^{135m}Xe	1.1×10^{-2}	7.38×10^{-4}	9.8×10^{9}	1.3×10^{13}
^{135}Xe	6.3×10^{-2}	2.10×10^{-5}	1.6×10^{9}	7.7×10^{13}
^{137}Xe	6.0×10^{-2}	3.02×10^{-3}	2.2×10^{11}	7.3×10^{11}
^{138}Xe	5.8×10^{-2}	8.15×10^{-4}	5.8×10^{10}	7.1×10^{13}
^{134}Cs	1.2×10^{-7}	1.06×10^{-8}	$\sim 4. \times10^{4}$(e)	$\sim 1.5\times10^{12}$(e)
^{136}Cs	1.6×10^{-4}	6.17×10^{-7}	1.2×10^{5}	2.0×10^{11}
^{137}Cs	6.2×10^{-2}	7.30×10^{-10}	5.5×10^{4}	2.3×10^{12}
^{140}Ba	6.0×10^{-2}	6.26×10^{-7}	4.6×10^{7}	7.3×10^{13}
^{141}Ce	6.0×10^{-2}	2.48×10^{-7}	1.8×10^{7}	7.3×10^{13}
^{144}Ce	5.2×10^{-2}	2.82×10^{-8}	1.8×10^{6}	4.5×10^{13}

a Based on 100% uranium fission; actually there is an increasing fraction with time related to the fission of generated plutonium.

b Meek, M. E. and B. F. Rider, "Compilation of Fission Product Yields", General Electric, Vallecitos Nuclear Center Rept. NEDO-12154–1 (1974)

c Albenesius, E. L. and R. S. Ondrejcin, "Nuclear-Fission Produces Tritium", Nucleonics *18* (9), 199 (1960).

d Equilibrium with longer-lived parent is assumed.

e Mountain, J. E., L. E. Eckart and J. H. Leonard, "Survey of Individual Radionuclide Production in Water-Cooled Reactors", University of Cincinnati Rept. (1968).

Notes:

1. Generation rate = thermal power x $\frac{\text{fission rate}}{\text{MWt}}$ x use factor x y x λ =

$$1825 \text{ MWt} \times \frac{3.1 \times 10^{16} \text{ fission/s}}{\text{MWt}} \times 0.8 \times y \times \lambda \times \frac{\mu\text{Ci}}{3.7 \times 10^{4} \text{ dis/s}}$$

2. Use factor of 0.8 is average for 1970 and 1971
3. Accumulation = thermal power x $\frac{\text{fission rate}}{\text{MWt}}$ x use factor x y $(1-e^{-\lambda t})$

Appendix C.2

Calculated Fission Product Concentrations in Reactor Coolant Water

Fission Product	Amount in 0.02% of fuel elements, μCi	Leakage rate, μCi/s	Summed turn-over constants, s^{-1}	Concentration, μCi/ml
^{83m}Kr	1.4×10^{9}	9.2×10^{1}	1.0×10^{-4}	5.5×10^{-3}
^{85m}Kr	3.2×10^{9}	2.1×10^{2}	4.3×10^{-5}	3.0×10^{-2}
^{85}Kr	5.8×10^{7}	3.8	1.6×10^{-7}	1.5×10^{-1}
^{87}Kr	6.0×10^{9}	3.8×10^{2}	1.5×10^{-4}	1.6×10^{-2}
^{88}Kr	8.6×10^{9}	5.6×10^{2}	6.9×10^{-5}	5.1×10^{-2}
^{89}Kr	1.1×10^{10}	6.7×10^{2}	3.7×10^{-3}	1.1×10^{-2}
^{89}Sr	1.1×10^{10}	1.1×10^{-1}	2.8×10^{-5}	2.5×10^{-5}
^{90}Sr	4.4×10^{8}	4.4×10^{-3}	2.8×10^{-5}	9.8×10^{-7}
^{95}Zr	1.6×10^{10}	2.6×10^{-2}	2.8×10^{-5}	5.8×10^{-6}
^{95}Nb	3.4×10^{10}	5.4×10^{-2}	2.8×10^{-5}	1.2×10^{-5}
^{99}Mo	1.4×10^{10}	2.8×10^{1}	3.1×10^{-5}	5.6×10^{-3}
^{103}Ru	8.0×10^{9}	1.3×10^{-2}	2.9×10^{-5}	2.8×10^{-6}
^{124}Sb	7.3×10^{4}	1.2×10^{-7}	2.8×10^{-5}	2.7×10^{-11}
^{131}I	7.8×10^{9}	1.0×10^{2}	2.9×10^{-5}	2.2×10^{-2}
^{133}I	1.6×10^{10}	2.1×10^{2}	3.8×10^{-5}	3.5×10^{-2}
^{135}I	1.5×10^{10}	2.0×10^{2}	5.7×10^{-5}	2.2×10^{-2}
^{131m}Xe	1.1×10^{8}	7.2	8.3×10^{-7}	5.4×10^{-2}
^{133m}Xe	4.6×10^{8}	3.0×10^{1}	3.7×10^{-6}	5.0×10^{-2}
^{133}Xe	1.6×10^{10}	1.0×10^{3}	1.7×10^{-6}	3.8
^{135m}Xe	2.6×10^{9}	1.7×10^{2}	7.4×10^{-4}	1.5×10^{-3}
^{135}Xe	1.5×10^{10}	1.0×10^{3}	2.1×10^{-5}	2.9×10^{-1}
^{137}Xe	1.5×10^{10}	9.5×10^{2}	3.0×10^{-3}	2.0×10^{-3}
^{138}Xe	1.4×10^{10}	9.2×10^{2}	8.2×10^{-4}	7.1×10^{-3}
^{134}Cs	$\sim 3.0 \times 10^{8}$	~ 3.9	1.6×10^{-5}	1.5×10^{-3}
^{136}Cs	4.0×10^{7}	5.2×10^{-1}	1.6×10^{-5}	2.0×10^{-4}
^{137}Cs	4.7×10^{8}	6.1	1.6×10^{-5}	2.4×10^{-3}
^{140}Ba	1.5×10^{10}	1.5×10^{-1}	2.9×10^{-5}	3.2×10^{-5}
^{141}Ce	1.5×10^{10}	2.4×10^{-2}	2.9×10^{-5}	5.3×10^{-6}
^{144}Ce	9.0×10^{9}	1.4×10^{-2}	2.8×10^{-5}	3.1×10^{-6}

Notes:

1. Amounts are 2×10^{-4} x values in column 5 of Appendix C.1.

2. Leakage rates are amounts x the following escape rate coefficients:
   - 6.5×10^{-8} s^{-1} for Xe, Kr
   - 1.3×10^{-8} s^{-1} for I, Cs
   - 2.0×10^{-9} s^{-1} for Mo
   - 1.0×10^{-11} s^{-1} for Sr, Ba
   - 1.6×10^{-12} s^{-1} for all others

3. Summed turnover constants $= \lambda_{\text{water loss}} + \lambda_{\text{demineralizer}} + \lambda_{\text{decay}}$

   where $\lambda_{\text{water loss}} = 1.6 \times 10^{-7}$ s^{-1}

   $\lambda_{\text{demineralizer}} = 1.56 \times 10^{-5}$ s^{-1} for Cs

   0 s^{-1} for Xe, Kr

   2.82×10^{-5} s^{-1} for all others

   λ_{decay} are values in column 3 of Appendix C.1

4. Concentration = leakage rate/(1.6 x 10^8 x summed turnover constants) where 1.6 x 10^8 is the reactor coolant volume in ml.

5. The values of $\lambda_{demineralizer}$ are for a flow rate of 300 kg/min through a mixed bed demineralizer that has decontamination factors of 2 for cesium and 10 for all other ions. These decontamination factors and the escape rate coefficients are from reference 3 in Section 1.4.
6. Values are for 1825 MWt reactor after operation with 80 percent capacity factor for 500 days, with 0.020 percent of fuel leaking radioactivity.

Appendix C.3

Calculated Radionuclide Concentration in Steam Generator Water

Radionuclide	Concentration, C_s, μCi/ml July 24, 1970	 Nov. 20, 1970	 March 16, 1971
	from fuel		
^{89}Sr	7×10^{-8}	5×10^{-8}	5×10^{-7}
^{90}Sr	2×10^{-9}	2×10^{-8}	3×10^{-8}
^{95}Zr	9×10^{-9}	9×10^{-9}	2×10^{-7}
^{95}Nb	1×10^{-8}	2×10^{-9}	1×10^{-7}
^{99}Mo	---	8×10^{-8}	3×10^{-6}
^{103}Ru	$<7 \times 10^{-10}$	1×10^{-8}	1×10^{-7}
^{131}I	4×10^{-5}	2×10^{-6}	3×10^{-6}
^{133}I	---	5×10^{-5}	2×10^{-6}
^{135}I	—	1×10^{-5}	5×10^{-5}
^{134}Cs	$.5 \times 10^{-7}$	3×10^{-6}	9×10^{-6}
^{136}Cs	4×10^{-8}	7×10^{-7}	2×10^{-6}
^{137}Cs	6×10^{-7}	3×10^{-6}	9×10^{-6}
^{140}Ba	6×10^{-8}	3×10^{-8}	1×10^{-7}
^{141}Ce	$<6 \times 10^{-10}$	1×10^{-8}	2×10^{-7}
^{144}Ce	$<8 \times 10^{-10}$	$<2 \times 10^{-8}$	3×10^{-7}
^{239}Np	---	1×10^{-7}	3 x [illegible]
gross alpha	$<2 \times 10^{-11}$	2×10^{-10}	6×10^{-9}
	from activation of water, cladding, and construction materials		
^{14}C	7×10^{-8}	4×10^{-7}	--
^{24}Na	---	1×10^{-6}	2×10^{-6}
^{32}P	2×10^{-8}	9×10^{-6}	--
^{35}S	$<7 \times 10^{-10}$	1×10^{-8}	--
^{51}Cr	2×10^{-7}	1×10^{-7}	2×10^{-7}
^{54}Mn	1×10^{-5}	2×10^{-7}	2×10^{-8}
^{55}Fe	3×10^{-5}	2×10^{-6}	$<2 \times 10^{-9}$
^{59}Fe	3×10^{-7}	4×10^{-8}	$<2 \times 10^{-9}$
^{57}Co	1×10^{-7}	6×10^{-9}	$<5 \times 10^{-9}$
^{58}Co	3×10^{-5}	2×10^{-6}	8×10^{-7}
^{60}Co	4×10^{-6}	2×10^{-7}	7×10^{-9}
^{63}Ni	4×10^{-7}	—	--
^{124}Sb	1×10^{-8}	2×10^{-9}	$<1 \times 10^{-9}$
^{182}Ta	$<7 \times 10^{-10}$	2×10^{-9}	$<3 \times 10^{-9}$

Notes:

1. Calculated by $C_s = (C_r w_r)/(w_s + v_s \lambda_d)$

where

C_s = radionuclide concentration in steam generator water, μCi/ml

C_r = radionuclide concentration in reactor coolant water (Table 2.1), μCi/ml

w_r = rate of water leakage from reactor into secondary coolant water, kg/day

Appendix C.3 cont'd)

w_s = rate of water discharge (blowdown) from steam generators, kg/day
v_s = amount of water in steam generators (1.2×10^5 kg), kg
λ_d = radioactive decay constant (column 3, Appendix C.1, multiplied by 86,400 s/day), day^{-1}

2. The following values in kg/day were used in the calculation:

	July 24, 1970	Nov. 20, 1970	March 16, 1971
w_v	110	180	310
w_s	17,000	15,000	26,000

Appendix D.1

Estimation of Airborne Radioactivity in the Environment

The atmospheric diffusion model for the Haddam Neck site is derived from the following Gaussian dispersion equation (see *Connecticut Yankee Atomic Power Company, Facility Description and Safety Analysis, Vol. I*):

$$\frac{\chi\bar{u}}{Q} = \frac{1}{\pi \sigma_y \sigma_z}$$

where

χ = ground-level centerline concentration, Ci/m^3
$\bar{u}$ = average wind velocity, m/s
Q = source release rate, Ci/s
σ_y = crosswind plume standard deviation, m
σ_z = vertical plume standard deviation, m

The equation estimates the plume centerline concentration at ground level from a source released at ground level.

Meteorological observations were conducted at the site before building construction began for approximately a 15-month period from the winter of 1962. Wind speed and direction frequency data were provided by instruments mounted 1.6 m and 33 m above ground level. Temperatures were measured 7.6 m and 37 m above ground level to obtain the temperature difference for classifying atmospheric stability. By releasing chemical tracers and measuring their concentrations in air, σ_y was determined as a function of downwind distance. Values of σ_z were obtained from data published by Sutton. The values of σ_y and σ_z were subsequently modified to account for the additional dilution expected from the turbulent wake of the vapor container building located next to the primary vent stack.

The site diffusion model calculated by the station operator for distances up to 28 km is shown in Appendix D.2. The station staff considers the model conservative in that the tracer studies showed greater dispersion than predicted. The result of one tracer test during stable atmospheric conditions, which shows dispersion to be greater by factors of 4 to 7, depending on distance, is shown as the lower line in Appendix D.2.

Using the site model, annual relative concentrations (χ/Q) at significant locations in the environs were estimated by applying factors for the annual sector wind frequency and average wind velocity. Wind frequencies were obtained from the wind roses for 16 directions during stable, neutral, and unstable atmospheric conditions, given in the Facility Description and Safety Analysis report. Average wind speeds, obtained from the same report, were calculated from average wind speeds for each stability class normalized to the frequency of the stability types occurring during a given wind direction. Average wind speeds for the three stability classes are as follows:

Stability class	Stability frequency, percent	Average wind speed, m/s
unstable	34	1.6
neutral	16	6.0
stable	50	4.0

Estimated χ/Q values and the calculational data are as follows:

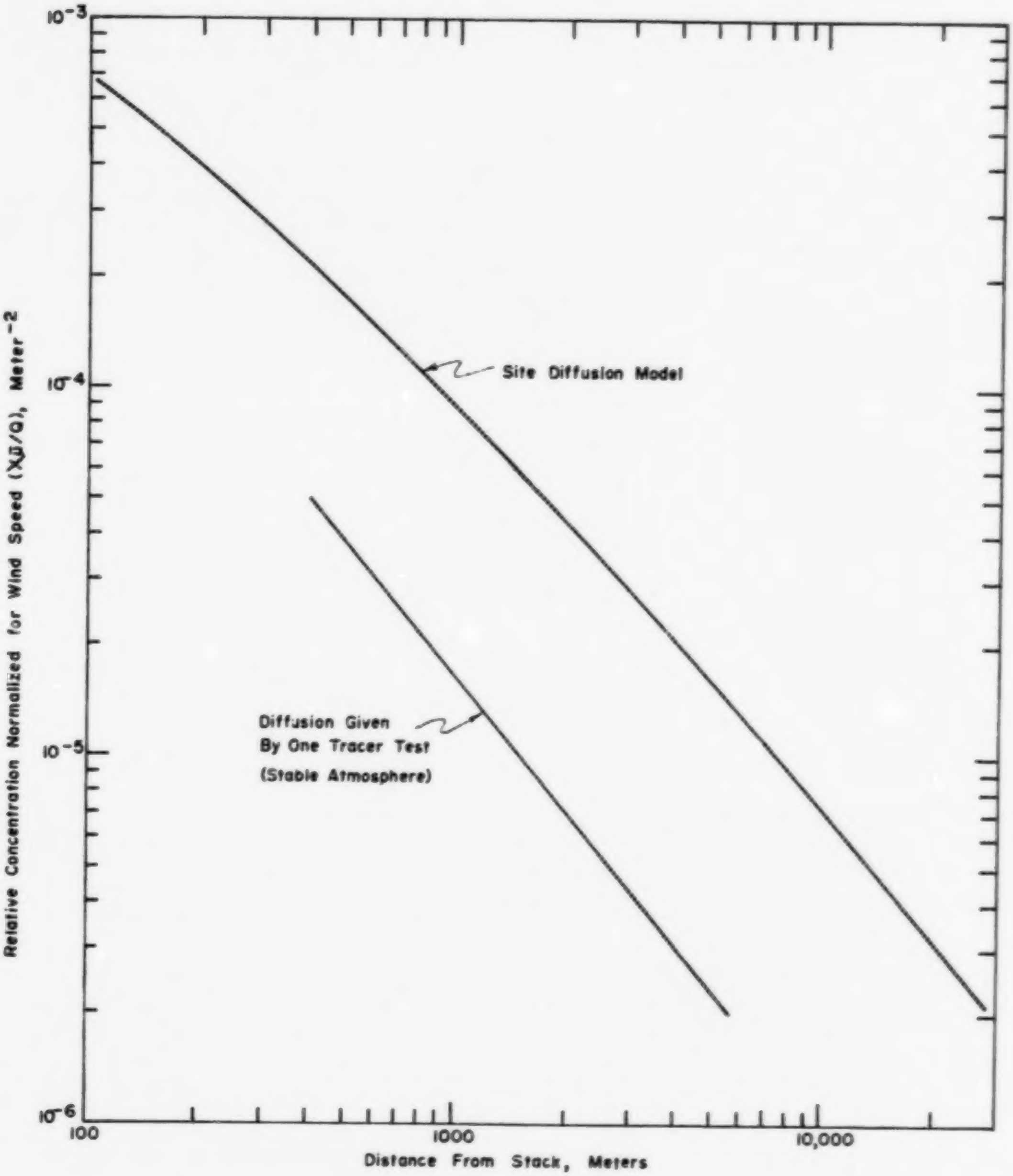
Appendix D.1 (cont'd)

Characteristic	Location	Relative concentration normalized to wind velocity $(\chi\bar{u}/Q)$, m^{-2}	Wind frequency %	Avg. wind velocity, $\bar{u}$, m/s	Avg. annual relative concentration $(\bar{\chi}/Q)$, s/m^3
Nearest residence	0.7 km WNW	1.2×10^{-4}	8.2	3.5	2.8×10^{-6}
Nearby residence	0.8 km NW	1.1×10^{-4}	6.6	3.2	2.3×10^{-6}
Nearby residence	0.8 km S	1.1×10^{-4}	2.6	3.9	7.3×10^{-7}
Nearby population of 70	1.2 km S	7.1×10^{-5}	2.6	3.9	4.7×10^{-7}
Nearby population of 89	1.2 km E	7.1×10^{-5}	6.9	3.1	1.6×10^{-6}
Nearby population of 46	1.2 km SSE	7.1×10^{-5}	4.5	3.5	9.1×10^{-7}
Nearest security fence	0.1 km NW	6.8×10^{-4}	6.6	3.2	1.4×10^{-5}
Nearest exclusion fence	0.5 km NW	1.7×10^{-4}	6.6	3.2	3.5×10^{-6}
Nearest dairy farm	3.4 km ESE	2.4×10^{-5}	13.2	3.5	9.1×10^{-7}
Fishing in canal	0.5 km SE	1.7×10^{-4}	15.6	3.6	7.4×10^{-6}
Fishing in canal	1.7 km SE	4.9×10^{-5}	15.6	3.6	2.1×10^{-6}
Fishing in river	0.2 km SW	4.0×10^{-4}	2.2	4.0	2.2×10^{-6}
State park	0.7 km SW	1.2×10^{-4}	2.2	4.0	6.6×10^{-7}

Values of $\bar{\chi}/Q$ calculated by the site diffusion model are about one-fourth as large as the annual average relative concentrations for various sectors and distances given in the *Haddam Neck Plant Environmental Report, Operating License Stage, Section 2.7 (1972).* On the other hand, estimates of $\bar{\chi}/Q$ in the Environmental Statement indicate less available dispersion, ranging from a factor of 11 at the fence to a factor of 2 at the dairy, than those given by the site model. The values for listed locations are as follows:

Location	$\bar{\chi}/Q$, s/m^3
0.1 km NW (nearest security fence)	1.6×10^{-4}
0.7 km WNW (nearest residence)	1.8×10^{-5}
3.4 km ESE (nearest dairy)	2×10^{-6}

Centerline concentration values at a receptor location overestimate annual average concentrations in a sector because the effluent is actually distributed across the sector. The average long-term concentration, χ_L (see equation 3.144 in *Meteorology and Atomic Energy 1968,* D. Slade, ed., USAEC Rept. TID-24190), for a 16-sector wind rose is related to the centerline value by $\chi_L/\chi = 6.38\sigma_y/x$, where x is distance from the stack. Use of σ_y values developed for the Haddam Neck site leads to ratios of χ_L/χ below unity at distances beyond 2.5 km from the source: at 3.4 km, the ratio is 0.9. Generalized σ_y values (*ibid.* pg. 102) weighted according to atmospheric stability frequencies given above yield a ratio of approximately 0.5 for distances of 0.1 to 3.4 km.



Appendix D.2 Atmospheric Diffusion Model for Haddam Neck Site, with Results from One Chemical Tracer Test*

* from Connecticut Yankee Atomic Power Company, Facility Description and Safety Analysis, Volume I

Appendix D.3

Atmospheric Dispersion and Plume Rise Estimates for Short-term Sampling

Plume dispersion during the tests described in Section 6 was estimated by the equation given in Appendix D.1, modified to take into account the elevation of the discharge from the stack:

$$\frac{\chi \bar{u}}{Q} = \frac{1}{\pi \ \sigma_y \ \sigma_z} \exp \left\{ -\frac{1}{2} \left(\frac{H}{\sigma_z}\right)^2 \right\}$$

where H = effective stack height ($53 + \Delta h$), m

Plume rise (Δh) was estimated by the ASME technique:

$$\Delta h = d \left\{ \frac{V_s}{\bar{u}} \right\}^{1.4}$$

where:

Δh = height of plume centerline above stack height, m

V_s = stack effluent velocity = 13.0 m/s

$\bar{u}$ = average wind speed at stack height, m/s

d = stack diameter = 1.8m

Parameters used to estimate dispersion during Test 3 (see Table 6.2) for slight atmospheric instability and 0.6 km distance to stack were:

Air sample number	$\bar{u}$ m/s	Δh, m	σy, m	σz, m
1	3.2	13	66	38
2	5.0	7	66	38
3	5.1	7	66	38
4	5.4	6	66	38

Appendix D.4

Estimated Deposition of ^{131}I at Farms Where Thyroids were Collected

Date		Wind frequency in 270°-360° sector,*(T)hr/day	Avg. wind speed,**($\bar{u}$) m/s	Deposition ^{131}I on ground †, (D), pCi/m^2
3.3 km ESE				
Aug.	25, 1970	9	3.09	$3.29 \times 10^{-4} Q'_o$
	26	3	1.55	2.19
	27	15	2.88	5.89
	28	12	1.80	7.53
	29	24	4.32	6.28
	30	6	1.80	3.77
	31	18	6.59	3.09
Sept.	1	21	4.86	4.88
	2	18	3.61	5.63
	3	9	1.55	6.56
	4	0	---	---
	5	12	4.51	3.01
	6	15	4.43	3.83
	7	12	3.73	3.64
	8	3	1.55	2.19
	9	0	---	---
	10	0	---	---
	11	21	4.34	---
	12	12	2.19	---
	13	0	---	---
	14	9††	4.29	2.37
3.3 km SE				
May	26, 1971	3	2.58	1.31
	27	12	3.35	4.05
	28	15	2.58	6.57
	29	3	1.55	2.19
	30	3	5.15	0.66
	31	9	3.61	2.82
June	1	12	2.96	4.58
	2	---	---	---
	3	---	---	---

* U. S. Dept. of Commerce, ESSA Local Climatological Data Sheets, Hartford, Conn., Aug. 1970, Sept. 1970, May 1971, June 1971.

** As reported from Hartford, for 90° sector. Divided in calculation by 4 to allow for 22.5° sector.

† For $(\chi\bar{u}/Q) = 2.5 \times 10^{-5}$ m^{-2}, $v_d = 10^{-2}$ m/sec,$(\chi_L/\chi) = 0.5$;
T = hrs. favorable wind direction in 270°-360° sector multiplied
by fraction (22.5°/90°) for smaller sector and by factor 3.6×10^3 s/hr.
Thus, $D = 1.13 \times 10^{-4} Q'_o (T/\bar{u})$

†† Deposition considered only for first half of day, since cow sent to market during this day.

Appendix D.5

Calculations of Estimated ^{131}I Levels in Cow Thyroids and Milk

Date		Deposition, $(10^{4} Q_0^{-1})^{-1}$, pCi/m^2 From Previous Day	Deposition Fresh Deposition	Deposition Total	Daily Intake $(10^{4}Q_0^{-1})^{-1}$ pCi/d	Decay Factor to Sept. 14	In thyroid, $(10^{4}Q_0^{-1})^{-1}$, pCi 20% daily uptake	In thyroid Cumulative amount to Sept. 14	Hypothetical conc. in milk % intake per liter	Hypothetical conc. in milk Concentration on Sept. 14 $(10^{4}Q_0^{-1})^{-1}$, pCi/l
Aug.	25	---	3.29	3.29	148	0.14	30	4		
	26	2.86	2.19	5.05	227	0.15	45	7		
	27	4.39	5.89	10.28	463	0.17	93	16		
	28	8.64	7.53	16.47	741	0.19	148	28		
	29	14.33	6.28	20.61	927	0.21	185	39		
	30	17.93	3.77	21.70	977	0.23	195	45		
	31	18.88	3.09	21.97	989	0.25	198	50		
Sept.	1	19.12	4.88	23.99	1080	0.28	216	60		
	2	20.87	5.63	26.50	1190	0.30	238	71		
	3	23.06	6.56	29.62	1330	0.34	266	90		
	4	25.77	---	25.77	1160	0.37	232	86		
	5	22.51	3.01	25.52	1150	0.41	230	94		
	6	22.20	3.83	26.03	1170	0.45	234	105		
	7	22.65	3.64	26.29	1183	0.50	237	119		
	8	22.87	2.19	25.06	1130	0.55	226	124	0.01	0.01
	9	21.80	---	21.80	981	0.61	196	120	0.01	0.01
	10	19.04	---	19.04	857	0.67	171	115	0.02	0.02
	11	16.54	---	16.54	744	0.74	149	110	0.04	0.04
	12	14.28	---	14.28	643	0.82	129	106	0.08	0.08
	13	12.53	---	12.53	564	0.91	113	103	0.22	0.23
	14	10.78	2.37	13.15	592	0.95*	118	112	0.35	0.39
Total								1604 = $0.16Q_0^{-1}$		= 0.78 x $10^{4}Q_0^{-1}$
3.3 km SE										
May	26	---	1.31	1.30	59	0.45	12	5		
	27	1.14	4.07	5.19	234	0.50	47	24		
	28	4.52	6.57	11.08	499	0.55	100	55	0.01	0.01
	29	9.64	2.20	11.83	532	0.61	106	65	0.01	0.01
	30	10.29	0.68	10.95	493	0.67	99	66	0.02	0.01
	31	9.53	2.80	12.35	556	0.74	111	82	0.04	0.03
June	1	10.74	4.58	15.32	689	0.82	138	113	0.08	0.09
	2	13.33	---	13.33	600	0.91	120	109	0.22	0.24
	3	11.64	---	11.64	524	1.00	105	105	0.35	0.37
Total								624 = $0.062Q_0^{-1}$		= 0.76 x $10^{4}Q_0^{-1}$

Note:

See Section 7.5 for factors (environmental half-life = 5 days, effective half-life in cow = 7 days, effective daily grazing area = 45m^2, ratio of ^{131}I concentration in milk to daily intake by cow from Curve "a" for sequential single intakes of ^{131}I in reference 13).

* For half day.

Appendix E.1

Relation of Airborne Radionuclide Concentration to Dose Rate

Radio-nuclide	Critical organ	Maximum permissible concentration per maximum permissible dose,(1) (pCi/cc) / (rem/yr)		
	Gases			
^{3}H (HTO)	Total body (In)(2)	2/5	=	0.4
(HT)	Skin (Sub)(3)	400/30	=	13
^{14}C (CO_2)	Fat (In)	1/5	=	0.2
	Total body (In)	2/5	=	0.4
^{41}Ar	Total body (Sub)	0.4/5	=	0.08
^{85m}Kr	Total body (Sub)	1/5	=	0.2
^{85}Kr	Total body (Sub)	3/5	=	0.6
^{87}Kr	Total body (Sub)	0.2/5	=	0.04
^{88}Kr	Total body (Sub)	0.28/5(4)	=	0.06
^{131m}Xe	Total body (Sub)	4/5	=	0.8
^{133m}Xe	Total body (Sub)	2.8/5(4)	=	0.6
^{133}Xe	Total body (Sub)	3/5	=	0.6
^{135}Xe	Total body (Sub)	1/5	=	0.2
other fission gases with half-lives < 2 hrs	Total body (Sub)	0.27/5(4)	=	0.05
	Airborne particles and iodine by inhalation			
^{54}Mn	Lung (I)(5)	0.01/15	=	0.00067
^{59}Fe	Spleen (S)(6)	0.3/15	=	0.020
	Lung (I)	0.3/15	=	0.020
^{58}Co	Lung (I)	0.02/15	=	0.0013
^{60}Co	Lung (I)	0.003/15	=	0.00020
^{89}Sr	Bone (S)	0.01/30	=	0.00033
	Lung (I)	0.01/15	=	0.00067
^{90}Sr	Bone (S)	0.0001/30	=	0.0000033
	Total body (S)	0.0003/5	=	0.000060
^{131}I	Thyroid (S)	0.003/30	=	0.0001
^{133}I	Thyroid (S)	0.01/30	=	0.00033
^{135}I	Thyroid (S)	0.04/30	=	0.0013
^{134}Cs	Lung (I)	0.004/15	=	0.00027
^{137}Cs	Lung (I)	0.005/15	=	0.00033

Appendix E.1 (cont'd)

1. ICRP, *Report of Committee 2 on Permissible Dose for Internal Radiation*, ICRP Publication 2, Pergamon Press, Oxford (1959); concentrations based on 168-hour limits.
2. (In) – Inhalation
3. (Sub) – Submersion
4. Based on ICRP Publication 2, equation 21, divided by 4 for a 168 hour week:

$$(MPC)a = \frac{2.6}{\Sigma(E)} \times 1/4 = pCi/cc,$$

where $\Sigma(E)$, the total effective energy per disintegration (β, γ, e^-, x-rays), has the values:

^{88}Kr = 2.33 MeV
^{133m}Xe = 0.234 MeV
Short-lived nuclides
($T_{1/2}$ < 2 hrs) = 2.42 MeV (based on ^{89}Kr, the radionuclide of highest disintegration energy with a half-life less than 2 hours)

5. (I) – Insoluble
6. (S) – Soluble

Appendix E.2

Relation of Daily Radionuclide Intake in Water to Dose Rate

Radionuclide	Critical organ	Maximum permissible intake per maximum permissible dose,(1) (pCi/day) / (rem/yr)		
^{3}H	Total body	66,000,000/5	=	13,200,000
^{14}C	Total body	17,600,000/5	=	3,520,000
^{24}Na	GI(LLI)	660,000/15	=	44,000
^{32}P	Bone	440,000/30	=	14,700
	Total body	1,980,000/5	=	396,000
	GI(LLI)	1,980,000/15	=	132,000
^{51}Cr	GI(LLI)	44,000,000/15	=	2,930,000
^{54}Mn	GI(LLI)	2,200,000/15	=	147,000
^{55}Fe	spleen	17,600,000/15	=	1,170,000
^{59}Fe	GI(LLI)	1,320,000/15	=	88,000
	Spleen	2,200,000/15	=	147,000
^{57}Co	GI(LLI)	11,000,000/15	=	733,000
^{58}Co	GI(LLI)	2,200,000/15	=	147,000
^{60}Co	GI(LLI)	1,100,000/15	=	73,300
^{90}Sr	Bone	200/0.167(2)	=	1,200
^{95}Zr	GI(LLI)	1,320,000/15	=	88,000
^{95}Nb	GI(LLI)	2,200,000/15	=	147,000
^{99}Mo	GI(LLI)	4,400,000/15	=	293,000
^{110m}Ag	GI(LLI)	660,000/15	=	44,000
^{131}I	Thyroid	80/0.500(2)	=	160
^{133}I	Thyroid	280/0.500(3)	=	560
^{135}I	Thyroid	800/0.500(4)	=	1,600
^{134}Cs	Total body	198,000/5	=	39,600
^{136}Cs	Total body	1,980,000/5	=	396,000
^{137}Cs	Total body	440,000/5	=	88,000

1. ICRP *Report of Committee 2 on Permissible Dose for Internal Radiation*, ICRP Publication 2, Pergamon Press, Oxford (1959): Intake based on 168-hour concentration limits.
2. based on recommendations by Federal Radiation Council in "Background Material for the Development of Radiation Protection Standards", Report No. 2, U. S. Government Printing Office, Washington, D. C. (1961).
3. based on ICRP, ^{133}I/^{131}I ratio: $80 \times 7 \times 10^{5}/2 \times 10^{5} = 280$.
4. based on ICRP, ^{135}I/^{131}I ratio: $80 \times 2 \times 10^{4}/2 \times 10^{5} = 800$.

Abstract

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A radiological surveillance study, one of a series at commercially operated nuclear power stations, was undertaken at the Haddam Neck (Connecticut Yankee) PWR plant. Radionuclide concentrations and external radiation were measured in the immediate vicinity of the 590-MWe station. The radionuclide contents and pathways of gases and liquids at the station, including points of discharge, were also measured to estimate radionuclide levels in the environment.

The radionuclides in airborne effluents were mostly ^{133}Xe, ^{3}H, and ^{85}Kr; shorter-lived noble gases, ^{131}I, and radioactive particles were also observed. The main discharges were from several sources, including the steam jet air ejector, ventilating air, and the gaseous waste system. Radioactive liquid effluents measured included those that originated in both the reactor primary and secondary coolant systems. Tritium was by far the major constituent.

Environmental radioactivity attributable to the station could be found most readily in the aquatic ecosystem, in water, fish, algae, plankton, plants, and sediment. Radioactive gases could be detected in ground level air only during releases from the waste surge sphere initiated specifically to test the detection equipment. One cow's thyroid contained a detectable amount of ^{131}I. Direct radiation from the station could only be measured on site. On the basis of effluent and environmental measurements, population radiation doses on the order of 1 mrem/year were estimated to occur by (1) fish consumption, (2) direct radiation, and (3) external radiation from gaseous effluents.

KEY WORDS:

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